UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 102 TO FACILITY OPERATING LICENSE NO. DPR-22

NORTHERN STATES POWER COMPANY

MONTICELLO NUCLEAR GENERATING PLANT

DOCKET NO. 50-263

1.0 OVERVIEW

1.1 Introduction

By letter dated July 26, 1996 (Ref. 1), the Northern States Power Company (NSP, or the licensee) requested an amendment to Facility Operating License No. DPR-22 for the Monticello Nuclear Generating Plant (MNGP). Specifically, the proposed amendment would increase the maximum licensed thermal power level by about 6.3 percent, from the current limit of 1670 megawatts thermal (MWt) to 1775 MWt, which is referred to as either "power uprate" or "power rerate." The amendment would also approve changes to the technical specifications (TSs) appended to the operating license to implement uprated power operation.

NSP later supplemented the original license amendment application by letter dated September 5, 1997 (Ref. 2), and revised the application in its entirety as described in a letter dated December 4, 1997 (Ref. 3). The revised application was then supplemented by letters dated March 6 (Ref. 4), March 26 (Ref. 5), April 8 (Ref. 6), April 17 (Ref. 7), and April 22 (Ref. 8), May 5 (Ref. 9), May 12 (Ref. 10), May 29 (Ref. 11), June 15 (Ref. 12), July 1 (Ref. 13), July 20 (Ref. 14), and July 30, 1998 (Ref. 15). The staff's initial proposed no significant hazards considerations determination (63 FR 9606) was based on the revised application dated December 4, 1997. The letters dated March 6, March 26, April 8, April 17, and April 22, May 5, May 12, May 29, June 15, July 1, July 20, and July 30, 1998, provided clarifying information that was within the scope of the original *Federal Register* notice and did not change the staff's initial proposed no significant hazards considerations determination.

1.2 Background

MNGP is currently licensed to operate at a maximum reactor power level of 1670 MWt. The licensee, in conjunction with General Electric Company (GE), undertook a program to uprate the maximum reactor power level by about 6.3 percent to 1775 MWt. At the uprated reactor power level, the generator electrical output will increase approximately 35 megawatts electric (MWe).

The licensee's plant-specific engineering evaluations supporting the power uprate were performed in accordance with guidance contained in the GE licensing topical report (LTR) NEDC-32424P, "Generic Guidelines for General Electric Boiling Water Reactor (BWR) Extended Power Uprate (ELTR1)" (Ref. 16). This topical report was previously reviewed and endorsed by the staff in a staff position paper dated February 8, 1996 (Ref. 17). For some items, bounding analyses and evaluations provided in GE licensing topical report, NEDC-32523P, "Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate (ELTR2)," (Ref. 18) were referenced. The staff has reviewed ELTR2 concurrently with the subject Monticello application, and the staff's safety evaluation on ELTR2 was issued on September 14, 1998 (Ref. 19).

The scope of the staff's review for the Monticello power uprate request was expanded since the date of the original power uprate amendment request (Ref. 1) and other guidance documents (Refs. 16 and 17) due to lessons learned from past power uprate amendment reviews. In particular, the licensee's request for power uprate was reviewed with consideration given to the recommendations from the Report of the Maine Yankee Lessons Learned Task Group. This report is documented in SECY-97-042, "Response to OIG Event Inquiry 96-04S Regarding Maine Yankee," dated February 18, 1997 (Ref. 20). The Task Group's main findings centered around the use and applicability of the computer codes and analytical methodologies used for power uprate evaluations. The Task Group also recommended that a standard review procedure for power uprate be developed to ensure that all appropriate review areas are addressed. For the BWR extended power uprate program, the staff had previously established, in Reference 17, review criteria and acceptable computer codes and analytical methodologies used for power uprate evaluations. In light of the Task Group's recommendations, the staff has expanded the review criteria to include areas such as human factors and offsite power stability. As a result, the staff concludes that the Maine Yankee Lessons Learned recommendations were appropriately considered in the review of the Monticello power uprate request.

1.3 Approach

The licensee's amendment request to uprate the current licensed reactor power level of 1670 MWt to a new limit of 1775 MWt represents an approximate 6.3 percent increase in thermal power with a corresponding increase in steam and feedwater flows of approximately 7.1 percent. The magnitude of the change in the reactor power level was chosen such that the maximum reactor vessel dome pressure and temperature and the maximum core flow rate remain unchanged at the uprated power level.

The planned approach to achieving the higher power level consists of (1) an increase in the core thermal power level to increase steam production in the reactor, (2) an increase in feedwater flow corresponding to the increase in steam flow, (3) no increase in maximum allowable core flow, and (4) operation of the reactor along extensions of current rod position/flow rate control lines. This approach is consistent with the generic BWR power uprate guidelines presented in Reference 16 and approved by the staff in Reference 17.

MNGP will increase core power by using a flatter radial power distribution while maintaining the most limiting fuel bundles within their operating constraints. The replacement of the high pressure turbine was designed to accommodate the increased steam flow at uprate conditions to assure a satisfactory pressure control margin without increasing the maximum operating dome pressure or turbine inlet pressure.

The licensee performed the accident analyses in support of the power uprate at 1880 MWt. This power level represents a bounding analysis, approximately 112.6 percent above the existing license limit of 1670 MWt; and approximately 6 percent above the MNGP proposed uprate power of 1775 MWt. The design-basis accidents were based on 102 percent of 1880 MWt, unless the 2-percent power factor was already accounted for in the analysis methods. Transient events were evaluated at a power level of 102 percent of 1775 MWt, unless the 2-percent power factor was already accounted for in the analysis methods. The initial operating dome pressure is assumed in the analyses to be 1040 psia, which is consistent with the current licensing basis used for MNGP.

All analyses were done for a representative fuel cycle with the reactor core operating at uprate conditions. Specific core performance is evaluated at each fuel cycle and will continue to be evaluated and documented for the fuel cycles that implement power uprate. Emergency core cooling system (ECCS) performance is evaluated at the bounding power conditions to show acceptable operation for power uprate. All limiting accidents and transients are analyzed based upon bounding conditions for power uprate to show compliance with regulatory requirements. The plant-specific evaluation for the MNGP is presented in the GE report, NEDC-32546P, "Power Rerate Safety Analysis Report for Monticello Nuclear Generating Plant," dated July 1996, as revised in Revision 1, dated December 1997. (Ref. 21).

The licensee also addressed the overall risk associated with the increase in the rated thermal power and concluded that no new vulnerabilities to severe accidents were found and that the change in CDF [core damage frequency] due to a bounding reactor thermal power increase of 12 percent is minor. The staff agrees with the licensee's conclusions as discussed in Section 5 of this safety evaluation.

2.0 EVALUATION OF SYSTEMS, STRUCTURES, AND COMPONENTS

The staff's review of the MNGP power uprate amendment request used applicable rules, regulatory guides, Standard Review Plan (SRP) sections, and NRC staff positions regarding the topics being evaluated. Additionally, the MNGP submittal was evaluated for compliance with the generic BWR power uprate program as defined in Reference 16. Detailed discussions of individual review topics follow.

2.1 Reactor Core and Fuel Performance

2.1.a Fuel Design and Operation

The effect of power uprate was evaluated for potential impact on various areas related to reactor thermal-hydraulic and neutronic performance. These included changes to the power/flow operating map, core stability, reactivity control, fuel design, control rod drives (CRDs), and scram performance. Additionally, the staff considered the impact of power uprate on reactor transients, anticipated transients without scram (ATWS), ECCS performance, and peak cladding temperature for design-basis accident break spectra. The power distribution in the core will be changed to achieve increased core power while limiting the absolute power in any individual bundle. Increased fuel enrichments or higher batch fractions may be used to provide additional operating flexibility.

Thermal-hydraulic design and operating limits assure an acceptably low probability of boiling transition occurring in the core anytime, even for the most severe postulated operational transients. Limits are also placed on fuel average planar linear heat generation rates to meet both peak cladding temperature (PCT) limits for the limiting loss-of-coolant-accident (LOCA) and fuel mechanical design bases. Subsequent core reloads at power uprate will also take into account these limits to assure acceptable margins between the licensing limits and their corresponding operating values. At power uprate conditions, all fuel and core design limits will continue to be met by control rod pattern adjustments. New fuel designs are not needed for power uprate to assure adequate safety. However, different fuel enrichment distributions may be used to provide additional operating flexibility and maintain cycle length.

2.1.b Thermal Limits Assessment

Fuel operating limits, such as the maximum average planer linear heat generation rate (MAPLHGR) and safety limit minimum critical power ratio (SLMCPR) for future reloads will continue to be met after power uprate. The methods used for calculation of MAPLHGR and operating limit minimum critical power ratio (OLMCPR) limits will not be changed because of power uprate, although the actual thermal limits may vary between cycles. Cycle specific thermal limits will be included in the Core Operating Limits Report. A representative cycle core is used for the uprate evaluation. These evaluations showed no change is required in the SLMCPR or the MAPLHGR and LHGR limits for power uprate.

2.1.c Reactivity Characteristics

All minimum shutdown margin requirements that apply to cold (212° F or less) conditions, will be maintained without change. Operation at higher power could reduce the excess reactivity during the cycle. This loss of reactivity is not expected to significantly degrade the ability to manage the power distribution through the cycle to achieve the uprated power level. The lower reactivity will result in an earlier all-rods-out condition. Any reduction in operational shutdown margins may need to be accommodated through core design. The technical specification requirements for shutdown margin will continue to be met.

(1) Power/Flow Operating Map

The power uprate flow map is shown in Figure 2-1 of NEDC-32546P (Ref. 21). Changes to the power/flow operating map are consistent with the generic descriptions given in Sections 5.2 and C.2.3 of NEDC-32424P (Ref. 16). The maximum thermal operating power and maximum core flow shown on Figure 2-1 correspond to the uprated power and the analyzed core flow range when rescaled so the uprated power is equal to 100 percent rated. Power uprate raises the upper portion of the core operating map (reactor power versus core flow) along the current rod/flow control lines. These lines have not changed but have been renamed to reflect the redefinition of rated thermal power. Full power operation under the maximum extended operating domain (MEOD), which was previously achieved at a minimum value of approximately 75 percent of maximum core flow, will now be achieved at approximately 81 percent of maximum core flow along the same rod lines. The absolute power MWt at that point on the operating map will be higher since the rated thermal power limit will be redefined. Present operational limits for the reactor recirculation system will constrain core flow to less than or equal to that achieved during power uprate operation. If these operational limits are changed in the future, a vibration analysis will be done to support operation within the licensed core flow limit.

2.1.d Stability

The BWR Owners' Group (BWROG) and the NRC have addressed methods to reduce the occurrence and potential effects of core power oscillations that have occasionally been observed at certain BWR operating conditions. Until the long-term stability option is incorporated into the TSs, the licensee has adopted the generic interim operating constraints proposed by GE (GE Service Information Letter SIL 380 Rev.1). Procedural requirements that restrict plant operation in the high power/low flow region of the power/flow operating map have been established in accordance with NRC Bulletin 88-07, "Power Oscillations in Boiling Water Reactors," and its Supplement 1. Plant operation after power uprate will extend the power/flow map to a higher power level (with corresponding higher flow), the current restricted operation regions of the power/flow map will remain unchanged, and operator actions upon entry into these regions remain the same. This is consistent with information presented in the generic evaluations provided by GE.

The licensee has selected the BWROG long-term stability option 1-D for the MNGP. TS changes in support of Option 1-D were submitted to the NRC, and the NRC approved the BWROG long-term stability Option 1-D for MNGP. This solution consists of an administratively controlled exclusion region and an analytical demonstration that, if an oscillation were to occur, (1) only core-wide mode oscillations will be expected, and (2) the existing flow-biased average power range monitor (APRM) flux trip will provide protection. The Option 1-D exclusion boundary is fuel- cycle dependent and represents a line of constant stability margins. The boundary is core power and flow dependent and is computed using the approved licensing procedure. Power uprate has been factored into the MNGP Option 1-D analysis so that the resulting exclusion region represents power uprate conditions. The Option 1-D analysis determined that regional mode oscillations remain improbable and that adequate safety limit MCPR protection continues for power uprate.

2.1.e Reactivity Control

The control rod drive (CRD) system was evaluated using the uprated steam flow and system pressure. The CRD system was evaluated for a reactor dome pressure of 1025 psia and a vessel bottom head temperature of 530 °F. These parameters were unchanged from the current licensed operating conditions. Increasing the reactor thermal power for power uprate will not affect the functions of the CRD system. Power uprate will result in a slight increase in the nominal operating dome pressure over that for current power operation (but still below the maximum operating pressure of 1025 psia). Higher reactor pressure assists the control rod during scram, resulting in a faster scram. The power uprate safety analysis did not take credit for the faster scram times and there is no change in the required scram time performance. The licensee has evaluated the CRD system for control rod insertion and withdrawal functions, plus CRD cooling, and concluded that the CRD system will continue to perform all of its functions at uprated conditions. The licensee will continue to monitor, through various plant TS surveillance requirements, the scram time performance to ensure that the original licensing bases for the CRD system are maintained. This approach is consistent with that proposed by GE in the generic references.

2.2 Reactor Coolant System and Connected Systems

2.2.a Nuclear System Pressure Relief

The purpose of the nuclear steam pressure relief system is to prevent overpressurization of the nuclear steam supply system (NSSS) during abnormal operational transients. In BWRs, the main steam line safety/relief valves (SRVs) provide this protection. Because the maximum operating dome pressure for power uprate is unchanged from that for current power operation, there are no changes to the nuclear system pressure relief system for power uprate. The nominal set points are unchanged from the set points for current power and the uprate analysis includes a 3-percent tolerance on the as-found opening pressure. The SRV operating pressure and temperature conditions for power uprate remain unchanged from current power operation because the reactor operating pressure remains unchanged. Therefore, the SRV design pressure and temperature are not affected by uprate. Previous plant operation with the current SRV set points has shown that an adequate difference exists between operating pressure and SRV set points (simmer margin) and that uprate operation should not result in an increase in inadvertent SRV actuation.

The licensee proposed changes to the bases for TS Sections 2.4, 3.6 and 4.6 that involve relaxation of the as-found setpoint tolerance for the SRVs from the current +2 percent to +3 percent at the power uprate condition. The nominal relief valve setpoint value remains unchanged. The bases of TS Section 2.4 specify that the operator will set the initiation lift setting at 1120 psig (1109 psig plus 1 percent) or lower. This initiation lift setting requirement provides assurance that the SRV set points do not drift beyond the allowable +3 percent tolerance limit.

In its September 5, 1997, submittal (Ref. 2), the licensee indicated that the proposed safety valve tolerance (+3 percent) from the nominal set points was incorporated into the abnormal transient and accident analyses at the power uprate conditions. The licensee also determined that peak pressure remains below the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code allowable of 110 percent of design pressure and that safety-related SRV operability is not affected by the proposed changes. MNGP has eight three-stage Target Rock SRVs. Industry operational reports indicated that, in general, these plant-specific model SRVs have not experienced significant setpoint drift. The licensee stated that the plant-specific analyses for the power uprate condition conservatively assume that only five of the eight plant SRVs are operable for the SRV safety function. This additional margin provided in the plant-specific analyses provides reasonable assurance that the SRV setpoint drift would not result in the maximum allowable system pressure being exceeded. Therefore, the staff agrees that the licensee's proposed as-found set pressure testing criterion for SRVs in the plant TS is acceptable.

2.2.b Reactor Overpressure Protection

The design pressure of the reactor vessel and reactor coolant pressure boundary will remain at 1250 psig after power uprate. The ASME Code allowable pressure limit for pressurization events is 1375 psig (110 percent of design value). Using the staff-approved model, the licensee analyzed the limiting pressurization event, which is a main steam isolation valve (MSIV) closure with failure of the reactor to scram automatically on MSIV position, which is described in Reference 18. Three SRVs were assumed to be out-of-service and an initial operating pressure of 1040 psia was used in the analysis. The analysis also assumed operation at 102 percent of 1775 MWt, 105 percent of rated core flow, a +3-percent tolerance on SRV opening pressure, and an automatic scram on high neutron flux during the event. At the uprated power, a peak pressure of 1287 psig resulted. This is higher than the present peak pressure but below the ASME Code's allowable limit. Therefore, the staff concludes that reactor overpressure protection will remain adequate after power uprate.

2.2.c Reactor Vessel and Internals

In Section 3.3 of Exhibit E (Ref. 1) and in the September 5, 1997, submittal (Ref. 2), NSP assessed the effects of the MNGP power uprate on the reactor pressure vessel (RPV) and reactor internals. Regarding the RPV, the licensee provided an assessment of (1) the impact of the uprate on the adjusted reference temperature (ART) of the limiting RPV material, (2) the need to revise the MNGP pressure-temperature (P-T) limit curves, (3) the change in the predicted upper shelf energy (USE) for the RPV materials and the validity of previously approved equivalent margins analyses, and (4) whether changes in the RPV surveillance program (as required by 10 CFR Part 50, Appendix H) are necessary. Regarding the reactor internals, the licensee provided an assessment of (1) changes in pressure differential loadings caused by the uprate, (2) changes in the assessment of flow-induced vibration caused by the uprate, and (3) changes in the potential for erosion damage due to the power uprate.

For analyzing the RPV, the licensee examined the effect of operating MNGP at a power of 1880 MWt until end-of-license (EOL) instead of the 1775 MWt that was being requested in the license amendment. The licensee's analysis therefore bounded the conditions expected from operation at 1775 MWt since RPV material embrittlement is directly related to the RPV neutron fluence, which is in turn related to the reactor operating power. In its September 5, 1997, submittal (Ref. 2), the licensee noted that the projected EOL neutron fluence 1/4 T location (i.e., at a point one-quarter of the way through the RPV wall from the RPV inside diameter) increased from 3.82×10^{18} n/cm^2 to 3.99×10^{18} n/cm^2 with the thermal power uprate from 1670 MWt to 1880 MWt.

In its December 4, 1997, submittal (Ref. 3), the licensee concluded that "comprehensive review...[of] the reactor vessel and internals....shows continued compliance with the original design and licensing criteria for the reactor vessel." The licensee went on to explain that with regard to the application of the requirements of 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," to the MNGP RPV materials:

(a) The upper shelf energy will remain greater than 50 ft-lb or [meets] the equivalent margin identified in [report NEDO 32205-A "10CFR50

Appendix G Equivalent Margins Analysis for Low Upper Shelf Energy in BWR/2 Through BWR/6 Vessels"] for the design life of the vessel.

(b) The beltline material reference temperature of the adjusted reference temperature (ART) will remain well within the 200 °F regulatory requirement.

(c) There is only a small change in the 32 effective full power year (EFPY) shift [in adjusted reference temperature], and consequently....[t]he existing P-T curves remain bounding for power operation up to a bounding 1880 MWt since they provide for adjusting the shift in RT_{NDT}.

Finally, in its September 5, 1997, submittal (Ref. 2), the licensee stated that "[n]o changes in the Appendix H program [the RPV surveillance program] are required because there are no significant changes in criteria important for Appendix G."

For the RPV internals, the licensee stated that the proposed power uprate will have no adverse effect on the flaw evaluations for existing flaws in the core shroud or in the core spray header. The licensee also concluded that the uprate would not affect the operation of any other reactor internals. The licensee's evaluation of the reactor coolant system piping confirmed that changes in the flow parameter associated with the power uprate would have no significant effects on the potential for flow-induced erosion/corrosion in those systems that might be susceptible to the phenomenon (i.e., the feedwater or main steam systems).

Regarding the RPV assessment, the staff has reviewed the information provided by the licensee and determined that the power uprate does not result in any significant change in the 10 CFR Part 50, Appendix G, upper shelf energy (USE) or P-T limit curve analyses for MNGP and does not necessitate any change in the MNGP 10 CFR Part 50, Appendix H, RPV surveillance program. Based upon the 1880 MWt 1/4T fluence values submitted by the licensee, the staff concluded that the lowest MNGP RPV material USE value at EOL (46.7 ft-lbs for RPV plates I-14 and I-17) submitted by the licensee was consistent with the results achieved by the staff using the Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2 methodology. This value reflected a USE reduction of 0.2 ft-lbs from the value calculated using the EOL fluence based upon operation at 1670 MWt.

In accordance with 10 CFR Part 50, Appendix G, RPV materials must maintain a USE of 50 ft-lbs throughout the life of the vessel or demonstrate that lower values provide margins of safety equivalent to those required by Appendix G to Section XI of the ASME Code. While the value of 46.7 ft-lbs is less than 50 ft-lbs, it remains well within the acceptable limits demonstrated in the NEDO-32205-A topical report on equivalent margins analyses for BWR/3 vessel materials. Therefore, the licensee has demonstrated that the MNGS RPV materials retain margins of safety equivalent to those required by ASME Code Section XI, Appendix G.

In evaluating the effect of the power uprate on the shift in limiting materials ART and the need for new P-T limit curves, the staff applied the methodology found in Regulatory Guide 1.99, Revision 2, for evaluation radiation embrittlement. This methodology specifies that a material's ART can be determined from the following equations:

$$ART = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin}$$

where, $\Delta RT_{NDT} = CF \times FF$

with, CF = the Chemistry Factor, a function of copper and nickel content

FF = the Fluence Factor, a function of the RPV material's neutron fluence

and, Margin = 34 °F (for all MNGP materials).

The staff identified that MNGP plate I-15 is still the limiting material based upon its initial reference temperature of 14 °F and its chemical composition (0.17 wt% Ni, 0.58 wt% Cu, CF = 125.3 °F). The change in fluence at the 1/4T location caused by the uprate (the point of interest for BWR P-T limit calculations) results in a change in the fluence factor from 0.734 to 0.745. Therefore, the ART at EOL of Plate I-15 changes from 140 °F to 141 °F. The staff has concluded that the format of the MNGP TS is such that this change can be accounted for and that no amendment of the MNGP TS is necessary. Also, the staff noted that while the criteria cited by the licensee (that the beltline material reference temperature of ART will remain well within the regulatory requirement of 200 °F) was previously part of a regulatory requirement, it was removed in the most recent revision to 10 CFR Part 50, Appendix G, and was therefore not relevant to the staff's evaluation.

Finally, based on the information cited above, the staff finds that no modification of the MNGP RPV surveillance program is necessary due to the power uprate. However, the staff notes that some concerns have been raised regarding BWR RPV surveillance programs in general with the Boiling Water Reactor Vessels and Internals Project (BWRVIP). These issues question whether certain BWRs possess unirradiated baseline surveillance data from which to measure changes in RPV material embrittlement and, if the data is not available, what actions can or should be taken to address the issue. MNGP has been one facility identified by the staff as potentially lacking baseline data for some surveillance program materials. While this topic does not directly affect the status of the facility's power uprate review, the staff expects that the licensee will participate in BWRVIP to address this outstanding issue.

The staff has reviewed the licensee's evaluations regarding the effect of the power uprate on core shroud and core spray piping and concludes that the licensee has bounded the effects of power uprate on the existing flaws. The staff concludes that the proposed power uprate will not affect the operation of core shroud, core spray header, or any other RPV internals. With regard to the RPV piping, the proposed power uprate will slightly increase the susceptibility to erosion/corrosion (E/C). In Section 3.6 of Reference 21, the licensee stated the following: "Appropriate changes to piping inspection frequency will be implemented to ensure adequate margin exists for those systems with changing process conditions. This action will take into consideration adjustments to predicted material loss rates used to project the need for maintenance/replacement prior to reaching minimum wall thickness requirements. This program will provide assurance that power uprate will have no adverse effect on high energy piping systems potentially susceptible to pipe wall thinning due to erosion/corrosion."

Based on the information presented above, the staff has concluded that RPV integrity, RPV internals, and reactor coolant system E/C issues have been adequately addressed to support plant operation at the power uprate conditions.

The licensee also evaluated the reactor vessel and internal components by considering load combinations that include reactor internal pressure difference (RIPD), LOCA, and seismic loads. The seismic loads are unaffected by the power uprate. The licensee recalculated RIPDs for the power uprate shown in Tables 3-1, 3-2, and 3-3 of Reference 21 for normal, upset, and faulted conditions, respectively.

The stresses and cumulative fatigue usage factors (CUFs) for reactor vessel components were evaluated by the licensee in accordance with the ASME Boiler and Pressure Vessel Code, Section III, 1965 Edition with addenda up to and including Summer 1966, to assure compliance with the code of record. The load combinations for normal, upset, and faulted conditions were considered in the evaluation. The maximum stresses for critical components were summarized in Table 17-1 of Reference 2 for the current operating and the power uprate conditions. The calculated stresses are less than the allowable code limits shown in the table. The calculated CUFs were provided in Table 3-4 of Reference 21 for limiting components such as refueling bellows skirt, closure region bolts and recirculation inlet nozzle.

In its September 5, 1997, submittal (Ref. 2), the licensee indicated that for cycle counting, actual transient cycles recorded at the plant are less severe than the assumed design-cycles used for calculating the fatigue usage factors. Therefore, the calculated CUFs that are based on design-cycles for the reactor vessel components at Monticello are considered conservative for the power uprate condition.

The licensee assessed the potential for flow-induced vibration based on the Monticello plant- specific vibration data for the reactor internal components recorded during the startup testing and operating experience from similar plants. The vibration levels calculated by extrapolating the recorded vibration data to power uprate conditions were found acceptable by the licensee in comparison to the acceptance limits specified in the Monticello Updated Safety Analysis Report (USAR).

Based on the above, the staff finds that the maximum stresses and fatigue usage factors as provided by the licensee are within the code-allowable limits and concludes that the reactor vessel and internal components will continue to maintain their structural integrity for the power uprate.

2.2.d <u>Control Rod Drive Mechanism</u>

The licensee indicated that the control rod drive mechanism (CRDM) has been designed in accordance with the code of record, the ASME Boiler and Pressure Vessel Code Section III, 1965 Edition with addenda up to and including Summer 1966. The components of the CRDM that form part of the primary pressure boundary, have been designed for a dome pressure of 1250 psig and a temperature of 575 °F, which is higher than the reactor bottom head pressure of 1045 psig, and temperature of 530 °F for normal and uprated power conditions.

The maximum calculated stress for the CRDM indicator tube is to be 20,790 psi which is less than the allowable stress limit of 26,060 psi. The analysis of cyclic operation of the control rod drive (CRD) resulted in a maximum CUF of 0.15 for the limiting CRD main flange for the power uprate. This is less than the code-allowable CUF limit of 1.0.

On the basis of its review, the staff concludes that the CRDM will continue to meet its design-basis and performance requirements at uprated power conditions.

2.2.e Reactor Recirculation System

The increase in reactor power will be accomplished by operation along extensions of current rod lines on the power/flow map with no increase in the maximum rated core flow. The licensee evaluated the cavitation protection interlock set point and concluded that it will remain the same in terms of absolute power. This interlock is based on subcooling in the external recirculation loop, which is primarily a function of flow, and thus is a function of absolute thermal power. The cavitation interlock is enforced in a region of the power/flow map that is not affected by power uprate. Therefore, the current cavitation interlock set point stays the same under power uprate conditions.

A small increase in flow resistance is expected to occur when operating at maximum core flow, due to an increase in the core average void fraction and a corresponding increase in two-phase flow resistance. The licensee estimated that the required pump head and pump flow at the uprate condition will increase the power demand of the recirculation motors by less than 2 percent for a given core flow. They concluded that the recirculation system can support power uprate operation.

The licensee evaluated the recirculation pump and jet pump net positive suction head (NPSH) for a bounding power of 1880 MWt and determined that the available NPSH is maintained greater than the required NPSH. Therefore, the licensee concluded that power uprate does not affect recirculation pump and jet pump NPSH.

In Section 10.4 (Ref. 21) on required testing the licensee stated no vibration testing of the reactor recirculation system will be necessary for power uprate. The recirculation pumps have been run at full speed to achieve increased core flow during an end-of-cycle coast down. Since the pumps have been run at full speed with acceptable vibration levels and system flows will not be increased for power uprate, the licensee stated that no additional vibration testing is necessary for power uprate. The licensee will continue to perform periodic surveillance tests, required in the plant TS to assure that the recirculation system will continue to meet its design and operating requirements at the increased maximum power conditions. The licensee should continue to monitor reactor recirculation pumps to assure that no undue vibration occurs at uprated power conditions. This is acceptable to the staff.

2.2.f Reactor Coolant Piping and Balance-of-Plant Piping

The licensee evaluated the effects of the power uprate condition, including higher flow rate, temperature, pressure, fluid transients, and vibration effects on the reactor coolant pressure boundary (RCPB) and the balance-of-plant (BOP) piping systems and components. The components evaluated included equipment nozzles, anchors, guides, penetrations, pumps, valves, flange connections, and pipe supports. The evaluation was performed using the original code of record specified in the Monticello Final Safety Analysis Report (FSAR), the code allowables, and analytical techniques. No new assumptions were introduced that were not in the original analyses.

The RCPB piping systems evaluated include main steam piping, reactor recirculation piping, feedwater piping, RPV bottom head drain line, reactor water clean-up (RWCU), reactor vessel head vent line, reactor core isolation cooling (RCIC), core spray piping, high pressure coolant injection piping (HPCI), residual heat removal (RHR), safety/relief valve (SRV) discharge piping and CRD piping. The licensee's evaluation of the RCPB piping systems consisted of computing the maximum increase in stress for the power uprate by comparing the increase in pressure, temperature and flow rate against the same parameters in the original design-basis analyses. The percentage increases in stresses for affected limiting piping systems were identified in Table 3-5 of the revised uprate license amendment request of Reference 21.

A majority of the RCPB systems were originally designed to maximum temperatures and pressures that bound the increased operating temperature and pressure due to the power uprate. For those systems whose design temperature and pressure did not envelop the uprate power conditions, the licensee performed stress analyses in accordance with the requirements of the code and the code addenda of record for the power uprate conditions. The licensee indicated that all safety-related piping was analyzed using the licensing basis computer codes. Plant piping was evaluated in accordance with ANSI B31.1, 1977 edition with addenda up to and including Winter 1978 to ensure compliance with the code of record. In its March 26, 1998, letter, the licensee indicated that the code for torus attached piping is ASME Code Section III, 1977 edition with addenda up to Winter 1978. The licensee concluded that the evaluation showed compliance with all appropriate Code requirements for the piping systems evaluated and that power uprate will not have an adverse effect on the reactor coolant piping system design.

The licensee evaluated the stress levels for BOP piping and supports in a manner similar to the evaluation of the RCPB piping and supports based on increases in temperature and pressure from the design basis analysis input. The evaluated BOP systems include lines that are affected by power uprate, but not evaluated in Section 3.6 of Reference 21, such as feedwater heater piping, main steam bypass lines, and portions of main steam, recirculation, feedwater, RCIC, HPCI, and RHR systems outside the primary containment. The limiting stress ratios of maximum calculated stresses to the allowable, resulting from the BOP piping evaluations for the power uprate, are shown in a table on page 15 of Reference 4. The staff finds that the stress ratios provided by the licensee are within the Code-allowable limits and are, therefore, acceptable.

The licensee evaluated pipe supports including anchorages, equipment nozzles, and penetrations by evaluating the effect of increased piping interface loads on the system components due to thermal expansion and on the margin in the original design basis. The licensee also performed detailed analyses using exact load combinations at the uprated conditions. The licensee indicated that all evaluated pipe supports were within the Code-allowable limits except a few supports that will require minor modifications. The affected piping supports include the RHR heat exchanger supporting base plates, one hanger on a feedwater heater drain line and several support modifications for the nonsafety-related main steam drain lines to the condenser. In Appendix H of Reference 21, the licensee committed to complete these piping support modifications prior to implementing the power uprate. Subsequently, in its letter dated July 30, 1998 (Ref. 15), the licensee stated that these piping support modifications have been completed. The effect of power uprate conditions on thermal and vibration displacement limits was also evaluated by the licensee for struts, springs and snubbers, and was found to be acceptable. The licensee reviewed the original postulated pipe break analysis and concluded that the existing pipe break locations were not affected by the power uprate, and no new pipe break locations were identified.

The licensee also evaluated the nonsafety-related main steam drain lines and condenser to ensure their structural integrity following a safe shutdown earthquake (SSE). The evaluation was necessitated by the projection of iodine deposition in the main steam lines and condenser for non-organic iodine released via the MSIVs following a DBA [design-basis accident]-LOCA. As such, to take credit for a holdup volume containing fission products, the licensee must demonstrate that the main steam drain lines and condenser remain structurally intact during and after an SSE. The staff's review of this issue is documented in Section 4.0 of this safety evaluation.

Regarding the assessment of the main steam flow restrictor, the licensee stated that there is no impact on the structural integrity of the restrictor for the power uprate. In Section 3.2 of the power uprate license amendment request, the licensee indicated that a higher peak RPV transient pressure (1287 psig) results from the Monticello plant operation at 1775 MWt conditions, but this value remains below the ASME code limit of 1375 psig. Therefore, the main steam line flow restrictor will maintain its structural integrity following the power uprate since the restrictor was designed for a differential pressure of 1375 psi which envelops the evaluated power uprate conditions.

Based on the above review, the staff concludes that the design of piping, components and their supports will be adequate to maintain the structural and pressure boundary integrity of the reactor coolant piping and supports in the power uprate conditions.

2.2.g Main Steam Isolation Valves

The performance of the MSIVs concerning RCPB requirements, such as closure time and leakage, could potentially be affected by the power uprate operating conditions. The MSIVs must close within specified limits at all design and operating conditions when receiving the signal to close. Existing design pressure and temperature for the MSIVs will bound the maximum operating pressure and temperature for power uprate conditions. MSIV performance will be monitored by surveillance requirements in the plant TSs to ensure that the original licensing basis for the MSIVs is preserved.

The MSIVs are designed to close fully within the time limits set forth in the plant USAR at maximum flow and maximum differential pressure conditions. These maximum conditions occur during a main steam line break event. The maximum flow rate and maximum differential pressure are dependent on maximum reactor dome pressure during the steam line break event and on the venturi design. Neither the maximum dome pressure nor the venturi design will be changed for power uprate, so the maximum steam flow rate and differential pressure will be unchanged. Therefore, the closure function and closure time will be unaffected. MSIVs are under scrutiny for leakage and closure time from surveillance requirements in the plant TS. The power uprate, therefore, does not affect the function of the MSIVs to perform containment isolation functions.

2.2.h Reactor Core Isolation Cooling

The RCIC system provides core cooling when the RPV is isolated from the main condenser, and RPV pressure is greater than the maximum allowable for initiation of a low pressure cooling system. The licensee has assessed the RCIC system consistent with the bases and conclusions of Section 3.1 of Reference 18 and concluded that the current RCIC system will maintain adequate water level for power uprate conditions. The licensee has evaluated the recommendation of GE SIL 377; specifically, to add a small bypass around the steam admission valve of the RCIC turbine to reduce the probability of a turbine overspeed trip during system startup. The evaluation concluded that adequate margins are provided to preclude a spurious trip of the RCIC system without the use of the GE SIL 377 recommendations. The maximum operating dome pressure and SRV opening set points will not be changed for power uprate, thus assuring the evaluated margins and reliability of the RCIC system. In addition, the RCIC control system uses a ramp generator during startup to provide controlled turbine acceleration. This feature provides optimum operation of the control system and minimum control valve cycling during system initiations. Therefore, the licensee determined that the use of the SIL 377 modification is not necessary for MNGP, thus providing assurance that operation at the power uprate will not degrade the reliability of the RCIC system.

2.2.i Residual Heat Removal System

The RHR system is designed to restore and maintain the coolant inventory in the reactor vessel and to provide decay heat removal following reactor shutdowns for both normal and post-accident conditions. The RHR is designed to operate in the low pressure coolant injection (LPCI) mode, shutdown cooling mode, suppression pool cooling mode, and containment spray cooling mode. The LPCI mode is discussed in Section 2.3.b of this safety evaluation.

The effect of power uprate on the shutdown cooling mode is to lengthen the time to reach the shutdown temperature (125 °F) for the primary coolant. The licensee estimates that the time to reduce the coolant temperature to 125 °F after steady-state operation at uprated power is still within the design objective of the RHR (to reach 125 °F within 24 hours using two RHR loops). It is assumed that the reactor coolant is cooled by steaming to the condenser to about 320 °F. At the uprated power level the decay heat is increased proportionately, thus increasing the time required to reach shutdown temperature. The licensee states that the RHR system at MNGP has the heat removal capacity to do this function up to the bounding reactor power of 1880 MWt. Thus, the RHR system still meets the original design bases with uprated power level.

The licensee also evaluated the alternate shutdown cooling path for a bounding reactor power of 1880 MWt as part of the 10 CFR Part 50, Appendix R, fire protection evaluation. The results of that analysis show that power uprate does not affect the alternate shutdown cooling analysis.

The effect of a bounding reactor power increase to 1880 MWt on the containment response is evaluated in Section 2.5. The analysis determined that the peak suppression pool temperature is 194 °F for the design-basis LOCA and 192 °F for the 10 CFR Part 50, Appendix R, event. These peak temperatures remain less than the ASME Code design limit of 281 °F for the containment and less than the torus attached piping analysis temperature of 195 °F. The staff's review of the licensee's analysis on the torus attached piping is also addressed in a separate safety evaluation, dated July 25, 1997 (Ref. 23).

2.3 Emergency Core Cooling System

Power uprate increases the calculated suppression pool temperature, which could decrease the NPSH available for the ECCS pumps. However, concurrent with the increase in suppression pool temperature, there is an increase in containment pressure that increases the NPSH available to the ECCS pumps. The licensee evaluated the containment response for NPSH at a bounding reactor power of 1880 MWt. This evaluation of containment response for containment pressure and suppression pool temperature was submitted in a separate license amendment request dated June 19, 1997 (Ref. 22), and was approved by a staff safety evaluation dated July 25, 1997 (Ref. 23).

2.3.a High Pressure Coolant Injection

The HPCI system design basis is to provide reactor vessel inventory makeup during small and intermediate break LOCAs and reactor vessel isolation events. The HPCI system is designed to provide its rated flow over a reactor pressure range of 150 psig to a maximum pressure based on the lowest SRV safety set point. The SRV opening set points will be not be increased for power uprate. The effect of power uprate on HPCI system operability, including potential system modifications, was addressed by GE in Reference 18. The power uprate operating conditions are bounded by the pump and turbine design conditions; therefore, power uprate should not affect the HPCI performance.

The licensee adopted the assessment of turbine overspeed as described in the generic topical report and has implemented GE SIL 480 for the HPCI system. The SIL 480 modifications have been carried out at MNGP to reduce the potential for a system trip during the startup transient. The continued reliability of the HPCI system at the power uprate condition will be monitored by the licensee as required by the Maintenance Rule (10 CFR 50.65).

2.3.b Low Pressure Coolant Injection

The hardware for the LPCI system is not affected by power uprate. The upper limits of the LPCI injection set points will not be changed for power uprate; therefore, the low pressure portion of the system will not experience any higher pressures. The design flow rates will not be changed. Since these systems do not experience different operating conditions due to power uprate, there is no effect due to power uprate. The adequacy of the LPCI system to provide core cooling during a LOCA is discussed in Section 2.4 of this safety evaluation.

2.3.c <u>Core Spray (CS) System</u>

The hardware for the CS system is not affected by power uprate. The upper limits of the CS injection set points will not be changed for power uprate; therefore, the low pressure portion of the system will not experience any higher pressures. The design flow rates will not be changed. Since these systems do not experience different operating conditions due to power uprate, there is no effect due to power uprate. The adequacy of the CS system to provide core cooling during a LOCA is discussed in Section 2.4 of this safety evaluation.

2.3.d <u>Automatic Depressurization System (ADS)</u>

The ADS uses SRVs to reduce the pressure following a small-break LOCA with high pressure ECCS failure. This function allows LPCI and CS to enter the vessel. The ECCS evaluation in Reference 21 shows that the initiation logic and ADS valve control are adequate for a bounding reactor power increase to 1880 MWt. Thus, the ADS remains acceptable for power uprate.

2.4 <u>ECCS Performance Evaluation</u>

The ECCS performance under all LOCA conditions and its analysis models must satisfy the acceptance criteria and requirements of 10 CFR 50.46 and 10 CFR Part 50, Appendix K. The results of the ECCS-LOCA analysis using NRC-approved methods are presented below.

A bounding ECCS-LOCA analysis was done at 102 percent of 1880 MWt, along with a benchmark analysis at 102 percent of 1670 MWt, using the NRC-approved SAFER/GESTR-LOCA model (Ref. 24). The results, which are summarized in Table 4-2 of NEDC-32546P (Ref. 21), show a negligible effect on peak cladding temperature (PCT). The PCT is determined primarily by the power in the peak bundle. For the ECCS-LOCA analysis the highest power rod in the peak bundle is assumed to be at the fuel rod peak linear heat generation rate (PLHGR); therefore, the peak bundle power does not change with power uprate. The relative difference between the peak bundle and the average bundle has a secondary effect on the PCT. The average bundle power increases in proportion to the core power increase, while the peak bundle remains constant. Therefore, the relative difference between the peak and average bundle powers decreases with power uprate. This affects the heat transfer and bundle reflooding for both the peak and average power bundles, resulting in an increase in the PCT for the average power bundle and a slight decrease in the PCT for the peak power bundle. The licensing basis PCT was 2087 °F at the power level of 1880 MWt, which is below the 10 CFR 50.46 PCT criterion of 2200 °F. The results of the ECCS performance evaluation show that the requirements of 10 CFR 50.46 and 10 CFR Part 50, Appendix K, are satisfied for a bounding reactor power to 1880 MWt, and thus power uprate for MNGP is acceptable. The ECCS evaluation also addressed power uprate in the maximum extended load line limit (MELLL) and increased core flow (ICF) regions of the power flow map. The analysis confirmed that the MAPLHGR multipliers remain acceptable for single loop operation in the MELLL region. The complete ECCS analysis was submitted in NEDC-32514P, "Monticello SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," dated July 1996 (Ref. 25).

The licensee resubmitted a LOCA analysis in Revision 1 to NEDC-32514P-Rev. 1 dated December 1997 (Ref. 26). This revised the previous analysis submitted in NEDC-32514P on July 26, 1996 (Ref. 25). The analysis was done with the NRC-approved SAFER/GESTR-LOCA licensing models. The SAFER and GESTR-LOCA models are coupled mechanistic, reactor system thermal-hydraulic, and fuel rod thermal-mechanical evaluation models. These models are based on realistic correlations and inputs. The calculation of the limiting PCT to show conformance with the requirements of 10 CFR 50.46 must include specific inputs documented in Appendix K to 10 CFR Part 50. The SAFER/GESTR-LOCA methodology requires:

- The licensing basis PCT (LBPCT) must be less than 2200 °F. This LBPCT is derived by adding appropriate margin for specific conservatism required by Appendix K to the limiting PCT value calculated using nominal inputs, the nominal PCT (NOMPCT).

- The upper bound PCT (UBPCT) must be less than the LBPCT. The UBPCT is the estimated mean of the PCT distribution for the limiting LOCA plus an adder (2 times the estimated standard deviation of the distribution of PCTs) for the limiting case LOCA. The UBPCT calculated in this way is presumed to bound the 95th percentile of the PCT distribution for the limiting case LOCA, and for all LOCAs within the design basis.

- It is required that the UBPCT be below 1600 °F; otherwise, additional plant-specific analyses must be done.

The specific analysis done for Monticello consisted of break sizes ranging from the smallest of 0.03 sq ft to the maximum DBA recirculation suction line break (4.095 sq ft). The break spectrum was evaluated with nominal and Appendix K analysis assumptions. The Appendix K PCT and the nominal PCT (NOMPCT) were compared to assure that the PCT trends as a function of break size were consistent with each other and with those of the generic BWR 3/4 break spectrum curves.

The MNGP plant-specific plant input operating parameters, fuel parameters, and ECCS parameters were used to perform the analysis. Various combinations of break locations, single failures, and available systems were also specifically analyzed for MNGP. The SAFER/GESTR-LOCA analyses were done with a bounding MAPLHGR at the limiting combination of power and exposure. The initial MCPR was based on a bundle power that is conservative with respect to the limiting bundle power expected during plant operation. The input parameters used are presented in Section 4 of NEDC-32514P-Rev. 1 (Ref. 26).

The results of the break spectrum calculations show that the NOMPCT increases with break size in the 1.0 sq ft to DBA range (consistent with the generic break spectrum trend). In this range the DBA recirculation suction line break with a LPCI IV failure results in the highest NOMPCT of 1180 °F. In the small break range (i.e.,< 1.0 sq ft), the PCT increases with decreasing break area. The 0.06 sq ft recirculation suction line break with battery failure results in a NOMPCT of 1295 °F. The MNGP nominal break spectrum shows the small break PCT is greater than the PCT for the DBA, whereas the generic small break PCT is lower than the PCT for the DBA.

The break spectrum was analyzed using the Appendix K input assumptions because the limiting break for MNGP is in the small break region. This established the Appendix K PCT versus break size curve to assure that the limiting break is consistent with the generic Appendix K results. The PCT for the largest recirculation suction line break with LPCI IV failure is 2029 °F. The limiting small break for the Appendix K case is 0.06 sq ft with a battery failure yielding a PCT of 1621 °F. The limiting fuel type with respect to both the nominal and Appendix K PCT in the MNGP analyses is the GE11 fuel for both large and small breaks. The PCT results for the break sizes analyzed for MNGP for both fuel types are in Section 5 of NEDC-32514P-Rev. 1 (Ref. 26).

The LBPCT for the limiting GE11 fuel with LPCI IV failure is 2087 °F, which is below the 10 CFR 50.46 limit of 2200 °F. The corresponding LBPCT for the GE10 fuel with LPCI IV is 1922 °F.

The UBPCT evaluation for Monticello followed the BWR 3/4 generic approach. This generic evaluation shows that the difference between the NOMPCT and the UBPCT is 349 °F, resulting in a UBPCT of 1380 °F. The 349° difference is composed of a generic modeling bias of 120 °F plus a plant variable uncertainty term of 229 °F. Since the Monticello PCT results are higher than the generic study, the plant variable uncertainty term was evaluated on a plant-specific basis. Break spectrum results showed that the limiting DBA PCT is the recirculation line break for the GE11 fuel. Plant-specific statistical analysis performed for the DBA-LOCA confirmed that the UBPCT remains below 1600 °F as required by the SAFER methodology. The UBPCT for the DBA-LOCA is less than 1550 °F for the GE11 fuel, and less than 1460 °F for the GE10 fuel. The maximum NOMPCT occurs for GE11 fuel at the 0.06 sq ft break size. An upper bound statistical analysis was performed for this break size and the results showed the UBPCT for the 0.06 sq ft break size is lower than the UBPCT for the DBA-LOCA, and is also less than 1600 °F. The UBPCT for the 0.06 sq. ft. break is less than 1480 °F for the GE11 fuel and less than 1420 °F for the GE10 fuel. This is below the LBPCT (2087 °F) as required by the methodology and meets the NRC acceptance criterion.

The LOCA analyses presented above were done to support the MNGP power uprate to a bounding thermal power level of 1880 MWt, corresponding to 112.6 percent of the current licensed power of 1670 MWt.

MNGP has implemented the maximum extended load line limit analysis (MELLLA) and ARTS [average power range monitor (APRM) system, the rod block monitor (RBM) system, and the associated technical specifications]. Both MELLLA and ARTS were previously approved by the staff. The higher rod line in the MELLLA region permits reactor operation at full power for core flows below rated. The PCT increased slightly over the 100-percent flow case, but remained well below the 10 CFR 50.46 limit. Flow-dependent MAPLHGR factors (multipliers) were derived for MNGP for operation at low flow conditions (less than 80 percent flow). These multipliers were derived based on the earlier, more conservative LOCA licensing evaluation model; therefore, the LOCA analysis for low flow conditions was not re-evaluated for MNGP with SAFER/GESTR.

The impact of ICF on LOCA limits was evaluated by the staff in a previously submitted SAFER/GESTR-LOCA analysis for MNGP for 105 percent rated core flow. A higher initial core flow provides increased core flow during the LOCA, increasing the margin to boiling transition. Other effects, such as change in core uncovery, are relatively small. The PCT for the worst break from the ICF initial condition is expected to be bounded by that obtained from the initial core flow at the rated condition. This conclusion should remain valid for power uprate.

Earlier evaluations for single-loop operation (SLO) imposed a multiplier on the MAPLHGR for the earlier and current fuel designs. These multipliers were derived to reduce bundle power levels such that the overall fuel heatup, which is increased due to an earlier boiling transition due to the reduction in initial core flow during SLO, is limited to acceptable values. The SAFER/GESTR-LOCA methodology provides for greater heat removal than the previous evaluation model; therefore, the use of the earlier multipliers is conservatively applicable for the SAFER/GESTR-LOCA analysis for SLO.

2.5 Containment System Performance

In accordance with the GE's topical report NEDC-32424P, "Generic Guidelines For General Electric Boiling Water Reactor Extended Power Uprate," (Ref. 16), the licensee has evaluated the effect of power uprate on the design- and licensing-basis of the containment system. These evaluations include containment pressure and temperature responses, containment dynamic loads, SRV containment dynamic loads, and subcompartment pressurization following a DBA-LOCA.

2.5.a Containment Pressure and Temperature Response

The current licensing-basis analyses for both the short- and long-term containment pressure and temperature responses following a large break LOCA are documented in the USAR. The short-term analysis is performed to determine the peak drywell pressure during the initial blowdown of the reactor vessel inventory into the containment, while the long-term analysis is performed to determine the peak pool temperature response considering the decay heat addition.

The licensee indicated that analyses were performed for the power uprate conditions in accordance with Regulatory Guide 1.49 ("Power Levels of Nuclear Power Plants") and NEDC-2424P, using GE codes and models. The M3CPT code was used (consistent with the current licensing-basis analyses) to model the short-term containment pressure and temperature response. The LAMB code was used for determining the RPV vessel break flow which is an input to the M3CPT code in the containment analyses.

The M3CPT break flow methodology treats the break flow in a simplistic and conservative manner. It assumes the pressure in the recirculation pipe remains constant until inventory in the recirculation loop is emptied (no flashing). The LAMB code more realistically models the recirculation loop as a separate pressure node and allows for inclusion of flashing in the pipe and vessel when the conditions warrant. A plant-specific calculation shows a pressure difference of approximately 5 percent in the peak drywell pressure at 1670 MWt using the M3CPT break flow with and without the LAMB blowdown model. The M3CPT code itself is used to calculate the drywell pressurization rate, vent clearing time, vent clearing pressure and associated loads. The staff has previously accepted the use of the LAMB code to model the RPV break flow in containment analyses for power uprate.

The licensee indicated that the SHEX code was used to model the long-term post-LOCA containment pressure and temperature response. The results of the benchmark analyses of the SHEX code to the HXSIZ code (the code used in the current licensing-basis analyses) at power levels of 1670 MWt and 1880 MWt were provided by the licensee. The benchmark analyses were performed using the May-Witt decay heat model and the ANS 5.1 nominal decay heat model. Using the May-Witt decay heat model, the peak suppression pool temperature was predicted to be 207.2 °F with the SHEX code and 207.6 °F with the M3CPT/HXSIZ code. Using the ANS 5.1 nominal decay heat model, the peak suppression pool temperature was predicted to be 193.6 °F with the SHEX code and 194.0 °F with the M3CPT/HXSIZ code. The results of the analyses demonstrated that the peak suppression pool temperature predicted with the SHEX code are within 1 °F of the peak pool temperature predicted with the M3CPT/HXSIX code. Based on the review of the benchmark analyses results, the staff finds the use of the SHEX code acceptable for MNGP power uprate analyses.

2.5.b Long-term Suppression Pool Cooling Temperature Response

(1) Bulk Pool Temperature

The licensee indicated that the long-term bulk suppression pool temperature response was evaluated for the DBA LOCA. A bounding analysis was performed at 102 percent of 1880 MWt using the SHEX code and the ANS 5.1 nominal decay heat model. The staff has determined that a 2σ adder (95 percent confidence interval) is necessary for the use of the ANS 5.1-1979 nominal decay heat model to account for the uncertainty. In a letter dated May 5, 1998 (Ref. 9), the licensee provided a comparative study between the generic 1880 MWt shutdown decay heat profile used for containment analyses and the MNGP-specific shutdown power profile for 1775 MWt with a 2σ adder. The comparative study shows that the nominal integrated energy at 1880 MWt bounds the integrated energy at 1775 MWt with the 2σ adder for the first 30 days post-LOCA. Therefore, it is reasonable to conclude that the generic 1880 MWt decay heat profile used in the power uprate containment analyses bounds the MNGP-specific 1775 MWt decay heat profile with the 2σ adder. Based on the above, the staff finds the bounding 1880 MWt nominal decay heat model acceptable for the proposed power uprate to 1775 MWt.

The licensee indicated that the long-term containment analysis was performed with the most limiting set of assumptions including the assumption of availability of containment cooling equipment (i.e., 1 RHR pump, 1 RHR service water pump, and 1 RHR exchanger) and the assumption of the maximum ultimate heat sink temperature. The use of the containment sprays was not assumed in this analysis. The analysis shows that, using the SHEX code and the ANS 5.1-1979 decay heat model as described above, the power uprate would increase the peak pool temperature by 8 °F, resulting in a DBA-LOCA peak suppression pool temperature of 194 °F. This is below the torus attached piping limit of 195 °F and the suppression chamber design temperature of 281 °F.

The licensee stated that the increased suppression pool temperature and pressure were analyzed for the potential impact on the NPSH for the ECCS pumps that draw water from the suppression pool. The NPSH analysis was performed for the bounding reactor power of 1880 MWt. The results of this analysis was provided by the licensee in a license amendment request dated June 19, 1997 (Ref. 22). The staff's review and approval of that license amendment request was documented in the staff safety evaluation dated July 25, 1997 (Ref. 23).

Based on the results of these analyses, the staff concludes that the peak bulk suppression pool temperature response remains acceptable for power uprate.

(2) Local Suppression Pool Temperature with SRV Discharge

A local pool temperature limit for SRV discharge is specified in NUREG-0783 ("Suppression Pool Temperature Limits for BWR Containments") because of concerns resulting from unstable condensation observed at high pool temperatures in plants without the SRV discharge quenchers. Elimination of this limit for plants with the quenchers on the SRV discharge lines is justified in GE report NEDO-30832, "Elimination of Limit on Local Suppression Pool Temperature for SRV Discharge with Quenchers," issued in May 1995. MNGP has quenchers on the SRV discharge lines, but the MNGP-specific configuration was not explicitly addressed in NEDO-30832. Therefore, the licensee evaluated the local suppression pool temperature for the bounding case as documented in GE report NEDC-24387, "Monticello Nuclear Generating Plant Suppression Pool Temperature Response," issued in December 1981. The results of the analysis show that the local pool temperature is not sensitive to the initial reactor power, and the recommended maximum local pool temperature is not exceeded for bounding reactor power increases to 1880 MWt.

Based on the above, the staff concludes that the local pool temperature limit will remain acceptable for power uprate.

2.5.c <u>Containment Gas Temperature Response</u>

The licensee indicated that the steam line break analyses were performed at both the current power and the bounding 1880 MWt reactor power conditions. The peak drywell gas space temperature for the steam line break was calculated to be 331 °F for the bounding 1880 MWt reactor power conditions, an increase of 1 °F over the results of the current 1670 MWt power level. Additional analyses were performed to determine the drywell shell temperature. The results of these analyses showed that the peak drywell shell temperature for both the current and the bounding 1880 MWt reactor power condition will be 273 °F. This temperature is below the drywell shell design temperature of 281 °F.

The licensee stated that the suppression chamber gas space peak temperature was calculated assuming thermal equilibrium between the suppression pool and the suppression chamber gas space. The analysis for a bounding reactor power increase to 1880 MWt shows that the bulk suppression pool and gas space temperature will increase slightly after LOCA to 194 °F and would remain below a limit of 195 °F (for the torus attached piping analysis) and the torus design temperature of 281 °F. Thus, the containment analyses show no adverse effect on the containment structure design temperature. Based on its review, the staff concludes that the containment drywell and suppression gas temperature response will remain acceptable for power uprate.

2.5.d Short-term Containment Pressure Response

The licensee indicated that a short-term containment response analysis was performed to demonstrate that operation with uprated power would not result in exceeding the containment design pressure. The analysis was performed for a double ended guillotine break of a recirculation suction line, the most limiting DBA-LOCA for containment pressure, and covers the blowdown period during which the maximum drywell pressure and the maximum differential pressure between the drywell and wetwell occur. Bounding analyses were performed at 102 percent of 1880 MWt, using methods accepted during the Mark I Containment Long-Term Program. Break flow was calculated using a more detailed RPV model. The reanalysis predicted a maximum containment pressure of 39.5 psig, which remains below the containment design pressure of 56 psig for Monticello. Based on its review, the staff concludes that the containment pressure response following a postulated LOCA will remain acceptable for power uprate.

2.5.e Containment Dynamic Loads

(1) LOCA Containment Dynamic Loads

The generic guidelines in NEDC-32424P (Ref. 16) specify that the power uprate applicant should determine if the containment pressure, suppression pool temperature, and vent flow conditions calculated with the M3CPT code for the power uprate are bounded by the analytical or experimental conditions on which the previously analyzed LOCA dynamic loads are based. If the new conditions are within the range of conditions used to define the loads, then the LOCA dynamic loads are not affected by the power uprate and thus do not require further analysis.

The LOCA containment dynamic loads for the power uprate are based on the short-term LOCA analyses, which provide calculated values for the controlling parameters for the dynamic loads throughout the blowdown phase. The key parameters are the drywell and wetwell pressure, the vent flow rates, and the suppression pool temperature. The dynamic loads considered in the power uprate evaluations include pool swell, condensation oscillation, chugging, and vent thrust loads. The licensee indicated that the containment dynamic loads evaluation for a small-break accident (SBA) was performed assuming the use of containment sprays consistent with the expected operator actions in accordance with the emergency operating procedures.

The licensee stated that the short-term containment response conditions with uprated power are within the range of test conditions used to define the pool swell and condensation oscillation loads for the plant. Furthermore, the long-term response conditions with the power uprate for which chugging would occur are within the conditions used to define the chugging loads. For all break sizes evaluated, power uprate results in no significant change in the conditions that result in chugging loads. The vent thrust loads for bounding uprate power conditions are calculated to be less than the plant-specific values determined during the Mark I Containment Long-Term Program. Therefore, the LOCA dynamic loads for MNGP will not be affected by the power uprate. Based on the above review, the staff concludes that the LOCA containment dynamic loads will remain acceptable for power uprate.

(2) SRV Containment Dynamic Loads

The SRV containment dynamic loads are comprised of discharge line loads (SRVDL), suppression pool boundary pressure loads, and drag loads on submerged structures. The loads are influenced by the SRV opening set point pressure, the initial water leg height in the SRVDL, SRVDL geometry, and suppression pool geometry. None of these parameters, including SRV opening set point, are changed for power uprate relative to the initial SRV actuations for MNGP. Therefore, the power uprate will not impact the SRV load definition for the initial SRV actuations.

For subsequent actuations (second pops), the only additional parametric change with power uprate is the time between SRV actuations. A higher water level at the time of second pop will result in higher SRV loads. The licensee stated that the effect of power uprate on the SRV discharge line was evaluated. The evaluation showed that with the current SRV low-low set logic parameters, the SRVDL water level reestablishs equilibrium height well before the subsequent SRV actuations. Therefore, the power uprate will not affect the subsequent SRV actuation load definition, and the subsequent actuation loads for power uprate would remain acceptable.

Based on the above review, the staff concludes that the SRV containment dynamic loads will remain below their original design values for power uprate and therefore remain acceptable.

(3) Subcompartment Pressurization

The generic guidelines in NEDC-32424P (Ref. 16) specify that the break flow will be compared with the analytical or experimental basis for the LOCA subcompartment pressurization dynamic loads. If the calculated break flow conditions with power uprate are within the range of break flow conditions used to define the loads, subcompartment pressurization dynamic loads are not affected by power uprate. The licensee indicated that the break flow mass and energy release at power uprate conditions were evaluated and were found to remain bounded by the release at the current power conditions when the allowable core flow operating range is considered. Therefore, the loads on the biological shield wall will not increase as a result of power uprate operation. Based on the above review, the staff concludes that the subcompartment pressurization effects will remain acceptable for power uprate.

2.5.f <u>Containment Isolation</u>

The licensee indicated that the system designs for containment isolation would not be affected by the proposed power uprate. All safety-related motor-operated valves will be capable of performing their intended functions at uprate conditions in accordance with Generic Letter 89-10. For nonmotor-operated valves, the system designs for containment isolation are not affected by power uprate. The capability of the actuation devices to perform under power uprate conditions has been evaluated and determined to be acceptable for these valves. Based on the above, the staff finds that the operation of the plant at the uprated power level will not impact the containment isolation system.

2.6 Engineered Safety Features and Associated Support Systems

2.6.a Standby Gas Treatment System

The function of the standby gas treatment system (SGTS) is to reduce offsite dose rates following a DBA by reducing halogen and particulate concentrations in gases that are present in the secondary containment prior to their discharge to the environment. Doses due to radioactive releases are limited to within the criteria of 10 CFR Part 100 and 10 CFR Part 50, Appendix A, General Design Criterion 19. The capacity of the SGTS is designed to maintain the reactor building at a slightly negative pressure and to provide one air volume change per day in the reactor building. The licensee stated that the function of the SGTS and the design of the particulate and charcoal filtration systems would not be affected by power uprate. Also, the licensee stated that the total post-LOCA halogen loading increases slightly at plant conditions for a bounding reactor power of 1880 MWt, but remains within the original design capability of the filters.

Based on its review, the staff finds that plant operation at the power uprate level has an insignificant impact on the ability of the SGTS to meet its design objectives and concludes that it is acceptable.

2.6.b Post-LOCA Combustible Gas Control

The post-LOCA combustible gas control system (CGCS) includes hydrogen recombiners that are utilized following a LOCA to maintain the containment atmosphere in a noncombustible mixture. The licensee evaluated the effect of power uprate on the capability of the CGCS to satisfy the requirements of 10 CFR 50.44 for a bounding reactor power increase to 1880 MWt.

The evaluation concluded that post-accident hydrogen and oxygen generation rates would increase in proportion to the power level. The TS limit for containment oxygen concentration was used as an initial condition in the evaluation. The weight of zirconium contained in the fuel cladding for current or future core designs that would be the most limiting on the operation of the CGCS was also used in the evaluation. Results of the evaluation revealed that the CGCS would remain fully capable of maintaining a noncombustible containment atmosphere at uprate conditions using the current methodology in the USAR.

Another evaluation was performed that assumed a more limiting value of the metal-water reaction of the fuel cladding to a depth of 0.00023 inch in accordance with Regulatory Guide 1.7 ("Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident") rather than the 5 percent originally assumed in the USAR. As a result, the licensee found that the CGCS was fully capable of maintaining a noncombustible containment atmosphere at uprate conditions with the more limiting assumptions.

In addition, the licensee evaluated MNGP against NEDO-22155, "Generation and Mitigation of Combustible Gas Mixtures in Inerted BWR Mark I Containments," June 1982, and found that for all BWR Mark I containments (including MNGP), peak containment oxygen concentrations are maintained well below the combustible gas limits in RG 1.7 without requiring hydrogen recombiners. The licensee concluded that the operation of the CGCS is not necessary to control oxygen concentrations below the noncombustible criteria at power uprate conditions.

Based on its review, the staff finds that sufficient margin is available in the CGCS capacity to control post-LOCA combustible gas levels at power uprate conditions and concludes that it is acceptable.

2.6.c Control Room Atmosphere Control System

The control room atmosphere control system, which includes the control room ventilation-emergency filtration train (CRV-EFT) system, is designed to maintain the control room envelope at a slightly positive pressure relative to the outside atmosphere to minimize unfiltered inleakage of contaminated outside air into the control room following a LOCA. The licensee stated that the heat load in the control room and the toxic chemical study of control room habitability would not be affected by power uprate. The increase in the radioactive source term caused by operating at higher power level would result in a potential increase in the control room operator dose. The staff's review of the licensee's dose calculation is provided in Section 3.5 of this safety evaluation. A separate license amendment request (Ref. 27) was submitted by the licensee to establish TS limits consistent with the evaluation of the control room operator doses. Based on its review, the staff determined that plant operations of the main control room atmosphere control system at the power uprate conditions is acceptable as documented in the staff SE, dated August 28, 1998 (Ref. 35).

2.7 Instrumentation and Control

The set point changes for the identified instrumentation for the new power level are predicated on the assumption that the analytical limits used by the licensee are based on the application of approved design codes.

The following TS changes have been proposed by the licensee:

1. TS Table 3.1.1, Function 9, Turbine Condenser Low Vacuum -

   Limiting Trip setting has been changed from ≥23 in. Hg to ≥22 in. Hg

2. TS Table 3.2.1, Function 6a, Shutdown Cooling Supply Isolation, Reactor Pressure interlock -

   Trip setting has been changed from ≤75 psig at pump suction to ≤75 psig at Reactor Steam Dome.

3. TS Table 3.2.2, Function C3, Low Pressure Core Cooling Pump Discharge Pressure Interlock -

   Trip setting has been changed from ≤ 100 psig to [≥ 60 psig and ≤ 150 psig]

4. TS Table 3.2.1, Function 3a, Reactor Cleanup System, Low Reactor Water Level -

   Trip setting has been changed from ≥ 10.6" above the top of the active fuel to ≥ 7" annulus.

5. TS Table 3.1.1, Function 7, Reactor Low Water Level -

   Trip setting has been changed from ≥ 7 in. (6) to ≥ 7 in. (annulus) and note 6 was deleted. Note 6 states that 7" of water level instrumentation is 10'6" above the top of the active fuel at rated power.

6. TS Table 3.2.3, Function 3, Rod Block -

   For two loop operation, trip setting has been changed from ≤0.66W + 58% to ≤0.66W + 53.6%.

   For single loop operation, trip setting has been changed from ≤0.58(W-5.4) + 50% to ≤0.66(W-5.4) + 53.6%.

7. TS Section 2.3.A.1.a and 2.3.A.1.b APRM Scram -

   For two loop operation, trip setting has been changed from ≤0.66W + 70% to ≤0.66W + 65.6%.

   For single loop operation, trip setting has been changed from ≤0.58(W-5.4) + 62% to ≤0.66(W-5.4) + 65.6%.

In addition to the above changes, the licensee will implement new set points for the instrumentation that is listed in the TS as a percentage of flow or pressure, as the actual set point of these instruments will change although the percentage has not been changed. The licensee has identified this instrumentation as follows:

(a) Main steam line high flow ≤140% rated
(b) Automatic bypass of turbine control valve fast closure and turbine stop valve scram is effective below 30% thermal power as indicated by turbine first stage pressure.
(c) APRM flux scram trip setting shall be no greater than 120%.

The licensee has also revised the associated TS Bases to incorporate the changes to the TS. In addition to these changes, the licensee has made some editorial and administrative changes to the TS to incorporate values based on the new thermal power level.

The licensee's submittal of July 26, 1996 (Ref. 1), identified that GE Licensing Topical Report NEDC-31336, "General Electric Instrumentation Setpoint Methodology," dated October 1986, was used for the instrument set point calculations. The staff has previously accepted the NEDC-31336 for instrument set point calculations in a safety evaluation dated February 9, 1993, and found it acceptable for establishing new set points in power uprate applications.

By letters dated April 14, 1997, and February 11, 1998, the staff requested additional information regarding set point margins for the new thermal power level. The licensee in its letters dated September 5, 1997 (Ref. 2), and March 6, 1998 (Ref. 4), provided the requested information. The proposed set point changes resulting from the power uprate are intended to maintain the existing margins between operating conditions and the reactor trip set points and do not significantly increase the likelihood of a false trip or failure to trip upon demand. Therefore, the existing licensing basis is not affected by the set point changes to accommodate the power uprate.

Based on the above review and justifications, the staff concludes that the licensee's instrument set point methodology and the resulting set point changes incorporated in the TS for the power uprate are consistent with the Monticello licensing basis and are, therefore, acceptable.

2.8 Electrical Power and Auxiliary Systems

2.8.a Offsite Power Stability

MNGP has three sources of offsite power to the plant via three separate transformers: 2R, 1R, and 1AR. Power uprate has the greatest effect on the operation of the 1R transformer. The 2R source (345 kV system) has a large excess capacity such that the additional electrical loading associated with power uprate does not present challenges to its operability. In addition, 2R is equipped with automatic load tap changing mechanisms on its secondary windings to maintain voltage on connected buses within nominal values from no load to full load. The 1AR transformer scheme (which can be fed either from the 345 kV system or the 115 kV system) includes a load shed that leaves only the safety-related buses. Because the load increase due to uprate is confined to the non-safety buses, the capability of 1AR to supply safety loads is unaffected by the increased uprate loading on the nonsafety buses.

The 1R source (115 kV system), however, requires a modification to support power uprate due to additional electrical loading on the nonsafety buses. In addition, certain voltage limits required from the offsite transmission network have been changed to support 1R operation for the uprate conditions and also to account for transmission network (i.e., grid) limitations.

Currently, to assure operability of the 1R source, the licensee has administrative limits on minimum 115 kV system voltage. A minimum voltage requirement corresponds to that voltage sufficient to recover steady-state voltage above the degraded voltage reset point when either (1) the plant is on 1R and No. 10 autotransformer is out of service (i.e., "weak grid") or (2) when the plant is on 1R and No. 10 autotransformer is in service (i.e., "strong grid"). The 115kV system is tied to the 345 kV system via No. 10 autotransformer which is equipped with an automatic load tap changer to provide voltage regulation to the 115 kV system. Due to the modifications made to the offsite system since March 1985 and resulting modifications to the voltage modeling method, higher administrative limits have been determined to be needed and are currently being implemented, as an interim action, when the No. 10 transformer is out of service.

Modifications made to the offsite system since March 1985 include:

- A second offsite feed was added to the 1AR source from the 345 kV system.
- A line voltage regulator was installed on the feed to the 1AR primary windings.

- The #11 unit auxiliary transformer (a delayed access offsite source) was replaced with a full capacity circuit (i.e., an immediate access offsite source) consisting of 2RS and 2R transformers fed from the 345 kV system. In addition, this source includes automatic load tap change mechanisms on the 2R transformer secondary windings.

- The cooling tower loads were removed from the plant auxiliary electrical system, thus eliminating the need for shedding these loads to maintain adequate voltage during a LOCA.

Resulting modifications in the voltage modeling method include:

- Voltage studies for the 1AR source from the 115 kV and 345kV feeds include the presence of the line regulator.

- Voltage studies for the 2R source take credit for the 2R secondary load tap changer for maintaining required voltage to safely load over the no load to full load voltage operating range of the grid. Voltage studies do not assume operation of the tap changer to bring voltages back above the undervoltage relay reset point of 3975 V because of reduced grid voltage following grid contingencies or trip of the plant.

- The limiting scenario for voltage studies (i.e., start of a single core spray (CS) pump from full plant load) was replaced with the sequencing on of all ECCS equipment. With the minimum source voltage available from the transmission network and "weak grid" conditions, it must be shown that safety bus steady-state voltage recovers above the degraded voltage relay reset point of 3975 V following the sequenced loading of all ECCS equipment.

- Voltage modeling methodology was modified to more accurately project actual system behavior.

The above described changes required the elimination of all margin for future load growth from the minimum voltage calculation for the 1R transformer. The elimination of load growth margin allowed no room for additional uprate loads for the present 1R configuration and increased administrative limits on minimum 115 kV system voltage. Additional loading under uprate conditions would also, without modification, result in exceeding the derate MVA limit for the 1R transformer for its present tap setting.

Currently, the no load tap setting on the 1R source is set at the 115,000 V to 4160 V position with the following 115 kV system limits in place:

- 115 kV System Voltage, 1R in service - 118.8 kV to 122 kV

- 115 kV System Voltage, 1R in standby - 117.5 kV to 122 kV**

- No. 10 autotransformer LTC control band - 119 kV to 121 kV

** The 1R standby limits are based on a scenario in which a LOCA occurs concurrent with a loss of the 2R plant source. The automatic transfer to 1R due to loss of 2R results in a trip of both recirculation motor-generator sets, a trip of both circulating water pumps, and the re-energization of only one feed pump. The large reduction in load allows for lower voltage requirements from the 115 kV system for 1R operability.

During the March-April 1998 refueling outage, the licensee has (a) changed the 1R no load tap setting to the 112,125 V to 4160 V position, (b) changed the tap settings on some instrument AC transformers to avoid overvoltage concerns as a result of the change in the 1R tap setting, and (c) implemented the following new 115 kV system limits:

- 115 kV System Voltage (1R in service or in standby) - 116 kV to 121 kV
- No. 10 autotransformer LTC control band - 117 kV to 119 kV

The licensee has determined that the combined effects of the change to the 1R tap setting and the new 115 kV system voltage limits will (a) provide for operation at power uprate conditions by restoring load margin, (b) enable the 115 kV system to provide adequate voltage to 1R with or without No. 10 autotransformer in service, (c) enable operation of the 115 kV system without need for special administrative limits to assure 1R operability, and (d) not result in operating the 1R transformer above its derated MVA limit due to increased uprate loading.

Results from the licensee's power uprate load studies (i.e., minimum voltage calculations) for each of the three sources demonstrates that safety bus steady-state voltage recovers above the degraded voltage relay reset point of 3975 V following the sequencing on of all ECCS equipment.

Results from the licensee's grid stability calculations indicate that the transmission network performance is such that steady state post-contingency voltages remain within required limits for the following contingencies:

- Loss of the nuclear power generating unit
- Loss of the largest generating unit
- Loss of the largest transmission line or intertie
- Loss of the largest system load

Results from the licensee's calculation CA-97-219, "Effects of Transmission System Performance on Offsite Source Operability," demonstrate that offsite system voltages stay within load study limits for offsite source operability subsequent to trip of the plant due to LOCA assuming (1) uprate power levels, and (2) pre-existing steady-state transient voltage levels occurring after any one of the following identified limiting expected transmission system lineups (i.e., limiting transients):

- Outage of the largest generating unit

- Outage of Substation Autotransformer No. 10.
- Outage of the primary 345 kV connection between the MNGP substation and the Twin Cities metropolitan area.
- Outage of the 345 kV connection between the MNGP substation and the 2340 MW Sherburne County generating facility
- Outage of two of the four 345 kV transmission lines interconnecting the MNGP and Sherburne County with the Twin Cities metropolitan area.

The licensee analyzed offsite source performance for uprate power levels and determined that all three offsite sources are operated within their ratings and remain capable of providing acceptable voltages to safety-related loads for limiting accidents and transients.

Based on a review of licensing/design-basis commitments documented in the Monticello USAR, it was initially the staff's understanding that operability of offsite sources required that the transmission network have sufficient capacity and capability (as demonstrated by stability analysis) so that acceptable voltage will remain available to safety system loads following simultaneous LOCA and any single failure on the offsite system or transmission network.

The licensee, in response to a request for confirmation of the above understanding, disagreed and indicated, by letter dated March 6, 1998 (Ref. 4), that (1) the original design basis for offsite/power systems is based on compliance with Principal Design Criterion 1.2.6, as described in USAR Section 1.2.6, and with its stated position on the proposed AEC General Design Criteria (GDC) Criterion 39, as described in Appendix E of the USAR page 34, (2) these design criteria (i.e., Principal Design Criterion 1.2.6 and the licensee's stated position) do not involve design-basis accidents coincident with network failures, and (3) the licensing basis does not require stability analyses for grid failures that occur simultaneously with a low probability LOCA which are not caused by the LOCA event. The following is a discussion of these issues:

Principal Design Criterion 1.2.6 defined in USAR Section 1.2.6, Plant Electrical Power

> Sufficient normal and standby auxiliary sources of electrical power are provided to attain prompt shutdown and continued maintenance of the plant in a safe condition under all credible circumstances. The capacity of the power sources is adequate to accomplish all required engineered safeguards functions under all postulated design-basis accident conditions.

The licensee's stated position as defined in Appendix E of the USAR Page 34

> Sufficient off-site and redundant, independent and testable standby auxiliary sources of electrical power are provided to attain prompt shutdown and continued maintenance of the plant in a safe condition under all credible circumstances. The capacity of the power sources are adequate to accomplish all required engineered safety features functions under all postulated design basis accident conditions. (Criterion 39)

Proposed AEC GDC Criterion 39 - Emergency Power for Engineered Safety Features (Category A) as defined in Appendix E of the USAR page 37

> Alternate power systems shall be provided and designed with adequate independency, redundancy, capacity, and testability to permit the functioning required of the engineered safety features. As a minimum, the on-site power system and the off-site power system shall each, independently, provide this capacity assuming a failure of a single active component in each power system.

During a March 30, 1998, meeting, the licensee further indicated that AEC GDC Criterion 39 was a proposed criterion at the time of Monticello licensing (a prelude to Criterion 17 of 10 CFR Part 50, Appendix A) and that the offsite system was not designed, nor was there a licensing/design-basis commitment for the offsite system to be designed, to meet the single-failure requirement specified in proposed Criterion 39. The licensee indicated that its offsite system design is required to meet Principal Design Criterion defined in USAR Section 1.2.6. Principal Design Criterion 1.2.6 requires sufficient offsite sources under all credible circumstances with adequate capacity to accomplish all required engineered safeguards functions under all postulated design-basis accident conditions.

Based on the results of the licensee's analysis, calculation CA-97-219 as described above, the staff concludes that the offsite power system will have adequate capacity, at the uprated conditions, to permit safety systems to perform their required function (after plant trip due to any postulated design-basis accident) from the offsite system's primary power source (normally the 2R transformer) following loss of the largest single supply to (or load from) the grid through a single normally anticipated transmission network event such as loss/trip of transmission line or generating unit. In addition, based on the results from the licensee's grid stability calculations, the staff concludes that the offsite power system will have adequate capacity either immediately (or on a delayed basis, if applicable), at the uprated conditions, to permit safety systems to perform their required function following a plant trip due to anticipated (nonaccident) contingencies at the plant or on the transmission network. In accordance with the license condition to update the USAR to reflect changes associated with power uprate, the licensee should incorporate CA-97-219 into USAR by reference.

Based on the above evaluation, the staff concludes that the offsite power system is consistent with Monticello's licensing basis which is Principal Design Criterion 1.2.6 and NSP's stated position in Appendix E of the USAR (defined above), and is, therefore, considered acceptable for the uprated conditions.

2.8.b Fuel Pool Cooling

The spent fuel pool cooling system (SFPCS) is designed to remove decay heat that is released from the spent fuel assemblies stored in the SFP, to maintain the SFP water temperature at or below its design temperature during plant operations, to reduce activity and maintain water clarity, and to maintain its cooling function during and after a seismic event.

As a result of plant operation at the uprated power level, the SFP heat load would slightly increase. The licensee evaluated the heat load in the SFP at the bounding 1880 MWt reactor power plant conditions to determine whether the SFPCS was able to maintain adequate fuel pool cooling. The licensee calculated the heat loads for planned refueling (normal heat load) and unplanned full core offload (emergency heat load).

The normal heat load analyzed at the bounding uprate level was determined to be 5.6 x 10^6 BTU/hr. This analysis was based on a partial core (141 assemblies) discharged every 18 months, with a cooling time of 96 hours (72 hours prior to discharge and a 24-hour discharge time) for the most recent refueling outage. The bulk pool temperature with river water (ultimate heat sink) at 90 °F was calculated to be 129.3 °F. One SFP cooling pump with flow through both heat exchangers was assumed. Based on the expected refueling cycle analysis, the licensee determined that the maximum heat load in the SFP would be less than the heat removal capability of the SFPCS heat exchangers. The licensee concluded that the SFP temperature would remain at or below the design temperature and adequate fuel pool cooling would be maintained for normal discharge conditions. The staff found that the bulk pool temperature was within the SRP 9.1.3 maximum pool temperature limit of 140 °F, and the 150 °F temperature limit specified in the American Concrete Institute Standard (ACI-349). The emergency heat load analyzed at the bounding uprate level was determined to be 20 x 10^6 BTU/hr. This analysis was based on a full core discharged 30 days after the last refueling discharge with the remaining spaces filled with spent fuel from previous refuelings. The analysis assumed that fuel pool cooling was provided by the RHR system, rather than the SFPCS, as documented in the USAR. The analysis also assumed a maximum SFP temperature of 140 °F, and then calculated the required RHR flow to maintain this temperature assuming the river water was at 90 °F. The licensee found that the calculated flows were all within the capability of the RHR system. The licensee concluded that power uprate would not have any significant effect on the capability to keep the SFP below the design temperature of 140 °F and to maintain adequate SFP cooling for all refueling scenarios. The staff found that the bulk pool temperature that was assumed in the analysis was within the SRP 9.1.3 criteria, and that the RHR capacity was sufficient to cool the SFP during emergency heat load conditions.

Based on the results of the licensee's analysis, the staff concludes that the slight increases in the SFP heat load and fuel pool temperatures due to power uprate would be within the design limits of the SFPCS, and are acceptable.

2.8.c <u>Cooling Water Systems</u>

The safety-related loads are rejected to the emergency service water (ESW) system, the residual heat removal service water (RHRSW) system, or the emergency diesel generator-emergency service water (EDG-ESW) system. The heat removed from these systems is rejected to the ultimate heat sink (UHS). The staff's evaluation of the effects of power uprate on each of these systems is provided below.

(1) Emergency Service Water System

The ESW system provides cooling water to the condensers for the control room ventilation air conditioning units, ECCS pump motor oil coolers, and RHR and HPCI room coolers under loss of offsite power and accident conditions. Heat is rejected to the ESW system by the RHR and core spray pump motors, RHR room cooler fan motors, ECCS sump pump motors, heat loss from piping, power cables, and room lighting, and heat transferred through room walls. The licensee stated that the increase in the calculated peak suppression pool temperature due to power uprate increases the piping heat losses and the heat transferred from the torus room through the walls to the RHR rooms. Based on equipment operability and environmental qualification requirements, the maximum long-term ambient temperature in the RHR room is 140° F. The licensee performed a bounding evaluation of the RHR room cooler performance during suppression pool heatup conditions for power uprate. The evaluation was approved by the NRC in a safety evaluation dated July 25, 1997 (Ref. 23). The licensee found that due to conservative assumptions in the evaluation and that the EDG load limits would restrict long-term ECCS pump operation to no more than two pumps, the RHR room coolers would maintain the RHR room below 140° F. Therefore, the licensee concluded that power uprate would not significantly affect the design function of the ESW system.

Based on its review, the staff finds that plant operation at the power uprate level has an insignificant effect on the design function of the ESW system and is acceptable.

(2) Residual Heat Removal Service Water System

The RHRSW system removes heat rejected by the RHR heat exchangers during normal reactor shutdown cooling, reactor isolation or accident conditions (suppression pool cooling), and when the SFP has an emergency heat load from the spent fuel. The system may also be used to supply makeup water to the SFP or the core, and can be used to flood primary containment. Heat loads for the shutdown cooling, suppression pool cooling, and emergency fuel pool cooling modes of operation are increased by power uprate. The licensee stated that for the limiting design-basis accident conditions, one RHRSW pump is sufficient to satisfy the minimum flow or makeup requirements at plant conditions for increases to a bounding reactor power of 1880 MWt for the auxiliary functions such as fuel pool makeup or core injection. The licensee concluded that the RHRSW system can support plant operation at the uprate power level without modification.

Based on its review, the staff finds that plant operation at the uprated condition has an insignificant impact on the design function of the RHRSW system and is acceptable.

(3) Emergency Diesel Generator- Emergency Service Water System

The EDG-ESW system removes the heat rejected by the EDGs when the EDGs are operating. The licensee stated that since the EDG loads remain unchanged for LOCA conditions following uprated power operation, the EDG-ESW can support plant operation up to a bounding reactor power level of 1880 MWt without modification. Based on its review, the staff agrees that power uprate has no effect on the function of the EDG-ESW system and is acceptable.

(4) Reactor Building Closed Cooling Water System

The RBCCW system is designed to remove heat during normal operation from the auxiliary plant equipment associated with the nuclear steam supply system housed in the reactor building. The licensee stated that the heat loads on the RBCCW would not increase significantly with power uprate since they depend mainly on either vessel temperature or flow rates in the systems cooled by RBCCW. The flow rates in the systems cooled by the RBCCW would not change due to power uprate. Therefore, the licensee concluded that power uprate has no significant effect on the RBCCW system function.

Based on its review, the staff finds that plant operation at the power uprate level has an insignificant impact on the RBCCW system and is acceptable.

(5) Ultimate Heat Sink

The UHS is the Mississippi River. The licensee stated that the effluent temperatures from the various SW systems would experience little or no increase as a result of the increased power levels. The circulating water system effluent temperature was calculated to increase about 2° F for reactor power operation at 1775 MWt, and about 4 °F when operating at the bounding 1880 MWt. The licensee concluded that the increases were minimal and would not affect the ability to meet NPDES [National Pollutant Discharge Elimination System] limits.

Based on its review, the staff finds that plant operation at the power uprate level has an insignificant impact on the ability of the UHS to meet its design objectives and is acceptable.

(6) Nonsafety-related Loads

The licensee stated that it has evaluated nonsafety-related cooling water systems, including the plant service water system and the main condenser/circulating water system, and determined that these systems are adequate to accommodate the increased heat loads associated with the plant operation at the uprated conditions.

Since these systems do not perform any safety function, the staff has not reviewed the impact of the proposed power uprate on the design and performance of these systems.

2.8.d Standby Liquid Control System (SLCS)

The function of the SLCS is to provide the capability of bringing the reactor from full power to a cold xenon-free shutdown assuming that none of the withdrawn control rods can be inserted. This function is met by the injection of a quantity of boron which produces an equivalent concentration of at least 660 ppm of natural boron in the reactor core in less than 125 minutes. SLCS shutdown capability (boron concentration) is reevaluated for each fuel reload to ensure sufficient shutdown margin is available. The SLCS performance was evaluated for a representative core design at a bounding reactor power of 1880 MWt. The results show that power uprate has no adverse effect on the capability of the SLCS to perform its function.

The SLCS is designed for injection at a maximum reactor pressure equal to the minimum SRV setpoint pressure. The nominal SRV set points and operating pressure will not be changed for the MNGP power uprate. The SLCS pumps are positive displacement pumps, where the small pressure increase related to the 3 percent tolerance on the as-found SRV opening pressure does not affect the rated flow to the reactor. Therefore, the capability of the SLCS to provide its backup shutdown function is not affected by power uprate. The SLCS performance is evaluated in Reference 21 for a representative core design at a bounding reactor power of 1880 MWt.

2.8.e Heating Ventilation and Air Conditioning (HVAC)

The HVAC systems consist mainly of heating, cooling supply, exhaust, and recirculation units in the turbine building, reactor building, and the radwaste building. The function of the HVAC systems is to prevent extreme thermal environmental conditions for personnel and equipment by ensuring that the design temperature limits are not exceeded. The licensee stated that heating and the ability to maintain minimum temperatures in any area of the plant during normal operation is not adversely impacted. Also, the ability of the system to minimize the spread of contamination is not adversely impacted since operation at uprate conditions does not change the flow path or differential pressure requirements of the HVAC system. The licensee stated that power uprate will affect those areas of HVAC that serve areas with condensate and feedwater lines or with lines that contain suppression pool water, which include the ECCS corner rooms and the steam chase. The licensee performed evaluations of the HVAC heat load increase due to a bounding reactor power increase to 1880 MWt and determined that the HVAC systems will accommodate the heat load increases due to power uprate without modification of the system.

Based on its review, the staff finds that plant operation at the power uprate level has an insignificant impact on the ability of the HVAC to meet its design objectives and is acceptable.

2.8.f Fire Protection

The licensee stated that the operation of the plant at the uprate power level does not affect the fire suppression systems, the fire detection systems, or the operator actions required to mitigate the consequences of a fire. There are no significant combustible load changes as a result of the power uprate. Also, the safe shutdown systems and equipment used to achieve and maintain cold shutdown conditions do not change and remain adequate for the power uprate conditions. The licensee evaluated the reactor and containment response to the postulated 10 CFR Part 50, Appendix R, fire event at a bounding reactor power of 1880 MWt and found that the peak fuel cladding temperature, reactor pressure, and containment pressures and temperatures were well below the acceptance limits. The licensee concluded that there is sufficient time for operators to take action to achieve and maintain cold shutdown conditions. Therefore, the fire protection systems and analyses are not significantly affected by power uprate.

Based on its review, the staff finds that the fire suppression and detection systems and their associated analyses are insignificantly impacted by power uprate, and concludes that the fire protection systems are acceptable.

2.8.g Systems Not Impacted by Power Uprate

In Section 6.8 of Reference 21, the licensee identified and evaluated plant systems that are affected in a very minor way by operation of the plant at the uprated power level. These systems were evaluated for bounding plant conditions for a reactor power of 1880 MWt. The licensee also identified, in Section 6.8 of Reference 21, systems that are not affected by plant operations at the power uprate level. Based on its review, the staff finds that plant operations at the proposed power uprate level have an insignificant or no impact on these systems.

2.9 Power Conversion Systems

The power conversion systems were designed to accept the steam and equipment flows resulting from continuous operation at 105 percent of rated throttle steam flow. At the time of MNGP's initial operation, the reactor core thermal power was "stretch" uprated to operate at 105 percent of rated throttle flow conditions. In order to accommodate a higher steam flow (approximately 7-percent increase) associated with the proposed power uprate, modifications to the turbine were required.

2.9.a Turbine-Generator

The power uprate operation will increase the steam flow by approximately 7 percent to achieve a reactor power of 1775 MWt. During the MNGP refueling outage in 1996, the high pressure (HP) turbine was modified with a turbine that was designed to support the 7-percent power uprate steam flow increase to 1775 MWt. The HP turbine modifications included the replacement of the HP turbine rotor and diaphragms. Also, the low pressure (LP) turbine was modified to replace the LP turbine rotors, inner casings, diaphragms, shaft packing boxes and steam guides. The licensee stated that the LP modifications were not necessary for power uprate, and would have been completed irrespective of power level. During the 1998 refueling outage, the first three rows of stationary diaphragms and the first row of buckets on the HP rotor were replaced to assure adequate steam flow capacity for power uprate operation. Also, the main generator stator cooling water skid was modified to enhance its cooling capacity for the uprated power. The licensee plans to test the flow passing capability to ensure the adequacy of the pressure control system during the power uprate implementation.

The licensee determined that power uprate does not involve an increase in the probability or consequences of a turbine missile event, and it does not change the conclusions in the turbine analysis in Section 12.2.3 of the MNGP USAR. The most limiting component in the turbine analysis is the LP turbine L-0 bucket, and the modifications for power uprate do not affect this component. In addition, turbine overspeed protection and the material strength of the LP buckets and rotors are not affected by power uprate. The turbine speed would remain at 1800 rpm, and power uprate would not involve sustained operation at critical turbine frequencies.

Based on its review, the staff found that the modifications that were performed on the turbine-generator do not change the conclusions in the existing turbine analysis that is documented in the MNGP USAR. Therefore, the staff concludes that operation of the modified turbine-generator at the power uprate level is acceptable.

2.9.b Miscellaneous Systems

The licensee evaluated the miscellaneous steam and power conversion systems and their associated components at the uprated power level. The systems include the condenser and steam jet air ejectors, the turbine steam bypass, and the feedwater and condensate systems. The licensee concluded that the existing equipment for these systems, with minimal modifications, are acceptable for operation at the power uprate level.

Since these systems do not perform any safety function, the staff has not reviewed the impact of the proposed power uprate on the design and performance of these systems.

2.10 High Energy Line Break Analyses

The licensee evaluated the high energy line breaks (HELBs) against the criteria set forth in the USAR for HELBs outside containment. The critical parameter affecting the HELB analysis for power uprate is an increase in reactor vessel dome pressure. Since there would not be an increase in the analyzed dome pressure, the licensee determined that the effect of power uprate on the HELB analysis was minimal. The slight changes in feedwater, condensate, and RWCU temperatures result in an insignificant increase in the mass and energy release rates following HELBs, which are within the margins in the existing environmental qualification envelopes. The licensee evaluated piping systems for the HELB, which include main steam, feedwater, HPCI, RCIC, RWCU, and steam jet air ejector. The licensee concluded that there were no new postulated break locations or pipe whip and jet impingement targets.

Based on its review, the staff finds that plant operation at the power uprate level has an insignificant impact on the HELB analysis and is acceptable.

2.11 Equipment Environmental Qualification

Environmental qualification (EQ) is based on composite HELB and design-basis accident/loss-of-coolant accident (DBA/LOCA) conditions and the resultant temperature, pressure, humidity, and radiation consequences. EQ also includes the environments expected during normal plant operation. For power uprate, the environment changes as follows:

- Inside containment, DBA/LOCA accident temperature and pressure environments will increase slightly.

- Outside containment, HELB accident temperature, pressure, and humidity environments will increase by an insignificant amount.

- For both inside and outside containment, radiation levels will increase proportionally (approximately 6.3 percent) with the increase in reactor power for normal and accident environments.

The licensee performed an evaluation comparing the power uprate environment to the existing equipment EQ documentation. For increased pressure, humidity, and radiation conditions, the licensee determined that the existing EQ test profiles continue to envelop uprate conditions.

For increased temperature conditions, the licensee determined that the existing EQ test profiles either continue to envelop uprate conditions or continue to "bound" the uprate conditions based on an equivalent aging/degradation analysis. The licensee, thus, concluded that equipment would remain qualified for power uprate conditions.

The licensee utilizes the Arrhenius methodology as the basis for its equivalent aging/degradation calculation (1) to demonstrate EQ for extended post-DBA operating times of 100 or 180 days using test data of a shorter duration and (2) to justify several locations where the EQ test temperature does not envelop the composite DBA power uprate profile. Transient temperature conditions are utilized to determine equivalent aging/degradation for test and expected (calculated) DBA conditions.

The use of the Arrhenius methodology to support qualification of equipment for DBA and/or longer term post-DBA environments has not been specifically endorsed by an NRC regulatory guide, has not been generally accepted, by itself, to demonstrate qualification of equipment in post-DBA environments, and has not been validated by test. Therefore, the staff has maintained that the use of Arrhenius methodology, by itself, without supporting justification or technical basis, is not considered an acceptable approach for supporting qualification of electric equipment for DBA environments.

The Arrhenius methodology is typically used for constant temperature conditions to simulate aging for generally constant normal environmental conditions prior to a DBA. The initial stages of a DBA composite temperature profile (a combined HELB and LOCA temperature profile) are anything but constant. However, towards the later stages of the DBA profile (after the first 4 to 5 days), the temperature drops to a relatively constant 200 °F if the profile is similar to IEEE-323 (Figure A1). For this relatively constant 200 °F temperature, the Arrhenius methodology can be considered acceptable provided the transient portion of the DBA composite profile is appropriately applied to (or excluded from) the Arrhenius calculation for extending qualification out to 100 or 180 days.

The staff is aware that the Arrhenius model does have several theoretical and practical limitations; for example: (1) reactions and aging mechanisms at high temperature and normal service temperatures may be different (i.e., activation energy may be a function of temperature, rather than constant); (2) the potential for problems when extrapolation through a material phase transition region (e.g., crystalline melting point region) is necessary; (3) use of regressed statistical data, which may generate varying results for activation energy based on the endpoint criterion selected; and (4) unavailability, in some instances, of activation energy information for specific material formulations (i.e., generally, activation energies are available for a given class of material). These limitations stem largely from the relatively simplistic kinetic model used as the basis for the relationship, and the use of empirical data for the determination of activation energy. Thus, the Arrhenius relationship should be used with caution. Although it may be used to provide a generalized description of the correlation between thermal exposure and degradation, it may produce results that are not representative of the actual behavior of the material under actual transient conditions.

Electric Power Research Institute's (EPRI's), Nuclear Power Plant Equipment Qualification Reference Manual indicates that the Arrhenius method has been employed to relate accident test temperatures to postulated accident temperatures. If the Arrhenius model and activation energy value are applicable to the test and accident temperatures, then the model may arguably be used in various ways to draw correlations between the accumulated thermal damage occurring during various phases of LOCA testing. This approach has been used principally to support long-term operability in post-LOCA environments when it is desirable to have a test duration that is shorter than the actual required operability time. For example, the test temperature plateau dropped to 212 °F at 5 days into the 30-day test. The required post-LOCA temperature dropped to 190 °F after 5 days and remained constant for an additional 175 days. Thus, although the test temperature envelops the required post-LOCA temperature, it lasts only 25 days and not 175 days. It is a common practice to argue that the higher test temperature (212 °F) can be viewed as an accelerated version of the actual post-LOCA temperature (190 °F). After using Arrhenius methods to determine equivalent degradation for 25 days at 212 °F and 175 days at 190 °F, if it turns out the equivalent degradation for 25 days at 212 °F is greater than 175 days at 190 °F, it can be argued that the test is conservative with respect to the actual post-LOCA conditions.

The staff agrees that the Arrhenius model and activation energy value can be shown to be applicable to the test and accident temperatures. The Arrhenius methodology has been typically used and accepted to project degradation (aging) under constant temperature conditions. In the example, the initial stage (e.g., the first 5 days) of the LOCA temperature profile, which is not constant, have been excluded from the Arrhenius calculation. The test conditions from 5 to 30 days and the accident profile from 5 to 180 days (175 days) for which the Arrhenius calculation is being applied are relatively constant. If temperature conditions are constant and if LOCA conditions do not cause material change, the staff agrees that the Arrhenius model and activation energy value can be shown to be applicable to support long-term operability in post-LOCA environments based on short-term testing at higher temperature.

As implied by the above described industry guidelines and discussion with others familiar with the application of the Arrhenius methodology, the staff has concluded that the Arrhenius model and activation energy value are generally not considered to be an accurate methodology for establishing degradation of equipment during transient temperature conditions, i.e., the initial stage of the DBA that is not constant. Thus, the staff disagreed with the licensee's application of the Arrhenius methodology during transient temperature conditions and its utilization to justify EQ where test temperatures do not envelop the conservatively calculated composite power uprate temperatures.

The licensee, in response to a supplemental request for additional information, provided the following supporting EQ information, by letter dated March 26, 1998, for each component/equipment type where expected conditions at the uprate power level exceed the conditions to which equipment was tested:

- Description showing relationship between uprate and test conditions,

- Evaluation demonstrating EQ where test conditions do not envelop uprate conditions, and

- Margins available at uprate power level derived through the use of the Arrhenius methodology utilizing transient temperature conditions.

Based on a review of this supporting EQ information, the staff concluded, except as noted below, that test conditions, for the most part, exceed expected power uprate conditions and that significant margins continue to exist at power uprate levels. Significant margins provide compensation for any uncertainties that may exist in the application of the Arrhenius methodology. In addition, the licensee's application of the Arrhenius methodology during transient temperature conditions does not appear to have a significant impact on qualification of equipment. Thus, except as noted below, the licensee's evaluation with supporting EQ information provides reasonable assurance that equipment required to meet the requirements of 10 CFR 50.49 will function as required during accident conditions with the higher uprate temperatures. However, as a separate initiative outside the scope of this evaluation, the NRC staff will continue to review this type of analytical methodology in order to assure that the approach used was appropriate and conservative.

For the following component/equipment types, EQ test temperatures do not envelop the composite DBA power uprate profile and compensating margin appears to be unavailable in the Arrhenius calculation to adequately account for uncertainties.

- For DG O'Brien Penetrations, test temperatures were identified to be below the composite temperature profile by about 25 °F from an average composite temperature of 225 °F for about 25 hours from about 100 minutes to 27 hours following a DBA.

- For General Atomic Rad Monitors, test temperatures were identified to be below the composite temperature profile by about 25 °F from an average composite temperature of 220 °F for about 23 hours from about 4 hours to 27 hours following a DBA and below the composite profile by about 10 °F from an average composite temperature of 150 °F for about 11 hours from about 36 to 47 hours following a DBA.

Staff concerns relating to uncertainties in the application of the Arrhenius calculation to justify EQ were discussed with the licensee during a March 30, 1998, meeting. The licensee, in response to these concerns, provided additional time versus temperature profiles by letter dated April 8, 1998. The new profiles used a linear versus log scale to better demonstrate the relative shape of the profiles on a common scale. The staff agrees with the licensee's assessment that these curves show that the areas bounded by the test curve have a much larger area than those areas not bounded by the test curve. However, these curves also clearly indicate that the Arrhenius methodology is being utilized at lower temperatures (for a much longer period of time) to justify EQ at higher temperatures (for a much shorter period of time). The staff feels that the utilization of a lower test temperature to justify EQ at a higher temperature (i.e., the reverse of how the Arrhenius methodology is normally applied) is an inappropriate application of the Arrhenius methodology. For example, testing at 200 °F for 48 (or 180) hours to justify EQ at a higher temperature of 220 °F for 24 hours (as is being proposed by the Monticello licensee) is considered an inappropriate application of the Arrhenius methodology. Arrhenius is normally applied at higher temperatures for short time periods to justify aging at lower temperature (i.e., normally expected temperatures) for a longer period of time. The staff, thus, concluded that EQ for DG O'Brien Penetrations and General Atomic

Radiation Monitors (based on the Arrhenius methodology) has not been appropriately justified in accordance with the requirements of paragraph (f)(1) of 10 CFR 50.49.

This conclusion, described above, was discussed with licensee as part of an April 17, 1998, conference call. As a result of this discussion, the licensee, by letter dated April 22, 1998, provided the following additional justification in support of its EQ for DG O'Brien Penetrations and General Atomic Radiation Monitors.

DG O'Brien Penetrations:

- The normal bulk ambient air temperature in the drywell is less than 135 °F.

- The DG O'Brien Penetrations were thermally aged prior to LOCA testing for 168 hours to a value of 249.8 °F. The accident profile over the area of concern is below 250 °F.

- The actual temperature response of the containment penetration, which is located at the containment wall, is expected to be lower than the predicted drywell air temperature response used for the accident profile.

- The temperature gradient applied to the penetrations would be lower than the predicted drywell air space temperature gradient (and the peak would also be lower) because the mass of the containment pressure boundary and the associated concrete shielding will act as a heat sink.

- The DG O'Brien Penetrations are uniquely positioned relative to the containment atmosphere. Each penetration is appended to an existing spare penetration. This spare penetration extends beyond the containment into the reactor building. The portion of the penetration that is environmentally qualified, the connector, is actually located in the reactor building.

- The drywell air temperature response is calculated using simultaneous application of worst-case conditions including an initial power assumption of 1917 MWt which is 8 percent above the requested uprate power level of 1775 MWt. The analytical containment temperature response profile provides a conservative bounding prediction of worst-case drywell temperatures with respect to the actual post-LOCA response.

- There are two DG O'Brien Penetrations in use at Monticello. These penetrations contain the instrument cables for the General Atomic Radiation Monitors. The radiation monitors were installed to meet the post-accident monitoring requirements of Section II.F.1.3 of NUREG-0737 ("Clarification of TMI Action Plan Items") and the guidelines of Revision 2 of Regulatory Guide 1.97 ("Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident"). These radiation monitors are not required to function to mitigate any design-basis accident. They have a post-accident monitoring function. They can also be used to detect increases in radiation levels above background during normal plant operations. They can also be used to determine gross estimates of fission product concentrations in the containment after a low-probability severe accident involving core damage.

- The radiation monitor instrument cables which utilize the penetrations are used for low signal strength applications, and current levels are very low. A failure of these cables would not involve a damaging arc. Given the uprate accident profile, the penetrations are unlikely to fail in a manner that would compromise containment integrity. Both the inner and outer pressure boundaries of the penetration would have to be compromised.

General Atomic Radiation Monitors

- These radiation monitors are required to meet the qualification guidelines of IEEE 323-1971. An NRC exemption was granted from the requirements of IEEE 323-1974.

- The radiation monitors were installed to meet the post-accident monitoring requirements of Section II.F.1.3 of NUREG-0737 and the guidelines of Revision 2 of Regulatory Guide 1.97. These radiation monitors are not required to function to mitigate any design-basis accident. They have a post-accident monitoring function. They can also be used to detect increases in radiation levels above background during normal plant operations. They can also be used to determine gross estimates of fission product concentrations in the containment after a low-probability severe accident involving core damage.

- For the area of the accident profile that is of concern, the temperature gradient is not limiting, and a failure of metal components is not indicated. The radiation monitor is constructed almost entirely of metal parts.

- The most susceptible component of the detector assembly is the nonmetallic Raychem cable splice. The splice material was separately tested, independent of the General Atomic qualification testing, to a longer and higher profile that bounds the post-LOCA containment temperature profile. In addition, the splice was thermally aged for 7318 hours at 276.8 °F and 830 hours at 323.6 °F.

Based on this additional supporting justification, the staff concluded that sufficient additional conservatism has been included in the EQ process and/or in the equipment's design to compensate for any uncertainties that may exist in the application of the Arrhenius methodology. In addition, the licensee's application of the Arrhenius methodology during transient temperature conditions does not appear to have a significant impact on qualification of equipment. The licensee's evaluation with this additional supporting EQ justification, thus, provides reasonable assurance that DG O'Brien Penetrations and General Atomic Radiation Monitors will function as required during accident conditions with higher uprate temperatures.

As noted above, the staff concludes that the licensee's evaluation with supporting information and justification provides reasonable assurance that equipment (required to meet 10 CFR 50.49) will be capable of performing its required function during a DBA when subjected to the higher power uprate conditions and is, therefore, considered qualified. In addition, based on the licensee's commitment that all affected environmental qualification files will be revised to reflect the new environmental profile changes associated with power uprate, the staff concludes that equipment (required to meet 10 CFR 50.49) will meet EQ requirements specified by 10 CFR 50.49 and is, thus, considered acceptably qualified for power uprate.

EQ of Mechanical Equipment with Nonmetallic Components

The licensee evaluated the effect of power uprate on the nonmetallic parts of equipment and components, such as pumps and heat exchangers. The MNGP Quality Assurance Program was reviewed for all structures, systems, and components (SSCs) that could be impacted by changes associated with power uprate. The licensee evaluated the changes in system pressures, temperatures, and flow rates and determined that most of the SSCs were within their original design capabilities with no additional actions needed. For the SSCs that require a design modification to assure compatibility with normal or accident conditions, these modifications were done under the MNGP quality assurance program. The quality assurance program provides controls to ensure that the design and qualification requirements are met.

Based on its review, the staff finds that the design and qualification requirements of nonmetallic parts of equipment and components would continue to be met for the uprated conditions.

2.12 Motor-Operated Valves (MOVs)

The power uprate at Monticello results in small changes in SRV setpoint tolerance, shutdown cooling isolation pressure, environmental temperatures, and electric supply power. With the resultant changes in the MOV performance requirements for its safety-related MOVs, the licensee determined that the RHR shutdown cooling valve MO-2030 requires an increase in its torque switch setting. In Exhibit D of its submittal dated December 4, 1997 (Ref. 21), the licensee committed to adjust the torque switch for MO-2030 to ensure adequate capability under the power uprate condition. The licensee also determined that HPCI steam line isolation valve MO-2034 and RBCCW to drywell isolation valve MO-4229 would close under the loaded conditions, but that their motors might not have sufficient capability to trip their torque switches under degraded voltage and ambient temperature conditions. In its initial evaluation in support of its power uprate request, the licensee determined that RHR discharge to torus valve MO-2006, RHR LPCI injection valve MO-2015, HPCI steam line isolation valve MO-2034, main steam line drain valve MO-2373, and RWCU inlet isolation valve MO-2398 would have small margins above the required capability for the power uprate conditions. By letter dated April 8, 1998 (Ref. 6), the licensee committed to initiate a condition report prior to operation at the uprate conditions to evaluate whether MO-2034 and MO-4229 have sufficient available capability for any subsequent operation after performing their safety functions. The licensee has performed modifications, adjustments, or more detailed calculations for MO-2006, MO-2015, and MO-2373 to provide additional margin. In its letter dated April 8, 1998 (Ref. 6), the licensee also committed to initiate a condition report to evaluate the capability margin for MO-2034 and MO-2398 prior to operation at the uprate conditions. For MO-2034, MO-2398, and MO-4229, the licensee stated that any corrective actions will be completed by the end of the next scheduled refueling outage. In its letter dated July 30, 1998 (Ref. 15), the licensee stated that it has initiated the above mentioned condition reports, and has adjusted the torque switch setting for MO-2030.

With respect to air-operated valve (AOV) capability under the power uprate conditions, the licensee reported that it had performed a qualitative review of systems containing safety-related AOVs. The licensee determined that no changes to system performance parameters will result from the power uprate that might affect safety-related AOV performance. The licensee stated that it is participating in the Joint Owners Group effort on AOV issues, but that a generic AOV program had not yet been established at Monticello.

The staff has reviewed the information provided by the licensee regarding the performance of its safety-related MOVs and AOVs under the plant conditions resulting from the Monticello power uprate. The staff finds that the licensee has adequately reviewed the capability of its safety-related MOVs and AOVs for the power uprate conditions. The staff notes that the licensee has made specific commitments (see Section 7.1.1 of this SE) to ensure the capability of several MOVs that might have low margins under the power uprate conditions.

2.13 Human Factors

The staff has reviewed the licensee's submittals dated July 26, 1996 (Ref. 1), September 5, 1997 (Ref. 2), December 4, 1997 (Ref. 3), and March 6, 1998 (Ref. 4), for power uprate. The staff's evaluation of the licensee's responses to the following five review topics is provided below.

Topic 1 - Discuss whether the power uprate will change the type and scope of plant emergency and abnormal operating procedures. Will the power uprate change the type, scope, and nature of operator actions needed for accident mitigation and will it require any new operator actions?

The licensee stated in its letter dated September 5, 1997 (Ref. 2), that the power uprate would not change reliance on symptom-based emergency operating procedures. The licensee also stated that the power uprate would not change the type, scope, or nature of operator actions needed for accident mitigation and that it would not require any new operator actions. The staff finds that the licensee's responses are satisfactory.

Topic 2- Provide examples of operator actions potentially sensitive to power uprate and address whether the power uprate will have any effect on operator reliability or performance. Identify operator actions that would necessitate reduced response times associated with a power uprate. Please specify the expected response times before the power uprate and the reduced response times. What have simulator observations shown relative to operator response times for operator actions that are potentially sensitive to power uprate? Please state why reduced operator response times are needed. Please state whether reduced time available to the operator due to the power uprate will significantly affect the operator's ability to complete manual actions in the times required.

By letter dated September 5, 1997 (Ref. 2), the licensee stated that the power uprate could decrease the time available for operator actions. Assuming a 12-percent bounding power uprate, rather than the proposed 6.3-percent power update, the licensee determined which operator actions would probably be sensitive to the power uprate. The licensee identified the following operator actions that would be most sensitive to the power uprate: (1) failure to manually depressurize the reactor vessel, and (2) failure to inject boron with the standby liquid control (SBLC) system.

The licensee reported the following: (1) about two-thirds of the increase in core damage frequency (CDF) due to the power uprate comes from failure to depressurize the reactor vessel, (2) the required time to initiate manual depressurization of the reactor vessel was changed from 26 minutes to 23 minutes, (3) about one-third of the increase in CDF due to the power uprate comes from failure to inject boron with the SBLC system during various ATWS scenarios, and (4) the time required to initiate SBLC changed from 21 minutes to about 13 minutes. By letter dated March 6, 1998 (Ref. 4), the licensee stated that the two actions (i.e., SBLC initiation and manual depressurization) are actions that are simple and ones that operators are frequently trained on. The licensee also stated that, for both actions, only switch manipulation from the control room is required: one switch manipulation for SBLC initiation and a maximum of three switch manipulations for manual depressurization. Further, the licensee stated that recent simulator observations for an operating crew demonstrated that these times can be achieved.

Based on the above, the staff finds that the reduction in the time available to the operator as a result of the power uprate should not significantly affect the operator's ability to complete the subject manual actions.

Topic 3 - Discuss any changes the power uprate will have on control room instruments, alarms, and displays. Are zone markings on meters changed (e.g., normal range, marginal range, and out-of-tolerance range)?

The licensee stated in its letter dated September 5, 1997 (Ref. 2), that the main control room can support the power uprate without modification and that no changes to zone markings have been identified. The staff finds that the licensee's responses are satisfactory.

Topic 4 - Discuss any changes the power uprate will have on the safety parameter display system (SPDS).

By letter dated September 5, 1997 (Ref. 2), the licensee stated that the changes to the SPDS due to the power uprate would be limited and principally pertaining to data validity checks. By letter dated December 4, 1997 (Ref. 3), the licensee committed to validate all affected SPDS data points. The staff finds that the licensee's commitment relative to validating SPDS data points is satisfactory.

Topic 5 - Describe any changes the power uprate will have on the operator training program and the plant simulator. Provide a copy of the post-modification test report (or test abstracts) to document and support the effectiveness of simulator changes as required by ANSI/ANS 3.5-1995, Section 5.4.1. Specifically, please propose a license condition and/or commitments that address the following:

(a) Provide classroom and simulator training on the power uprate modification.

The licensee stated in its letter dated September 5, 1997 (Ref. 2), that classroom and simulator training will be provided to all operations and licensed personnel in accordance with Monticello Training Center procedures. This training will be completed prior to implementation of the power uprate.

(b) Complete simulator changes that are consistent with ANSI/ANS 3.5-1985. Simulator fidelity will be re-validated in accordance with ANSI/ANS 3.5-1985, Section 5.4.1, "Simulator Performance Testing." Simulator re-validation will include comparison of individual simulated systems and components and integrated plant steady-state and transient performance with reference plant responses using similar startup test procedures.

The licensee stated in its letter dated September 5, 1997 (Ref. 2), that simulator changes will be completed in accordance with ANSI/ANS 3.5-1985, Section 5.4.1, simulator performance testing and Monticello simulator configuration control procedures. Initial simulator changes will be completed prior to power uprate and will be verified against actual plant startup data.

(c) Complete control room and plant process computer system changes as a result of the power uprate.

See evaluation of Topics 3 and 4, above.

(d) Modify training and plant simulator relative to issues and discrepancies identified during the startup testing program.

The licensee stated in its letter dated September 5, 1997 (Ref. 2), that the simulator will be modified in accordance with applicable Monticello Training Center procedures to reflect issues and discrepancies identified during startup.

The licensee also stated in its September 5, 1997, letter that the formal commitments to these conditions will be documented in its revised license amendment request. In Exhibit H, "License Commitments Associated with Rev. 1 of the MNGP Power Uprate License Amendment Request, " dated December 4, 1997 (Ref. 3), the licensee made the following commitments:

- Simulator changes will be completed in accordance with ANSI/ANS 3.5-1985 Section 5.4.1, simulator performance testing and Monticello Training Center procedures.
- Classroom and simulator training on new knowledge and abilities associated with the power uprate will be provided in accordance with Monticello Training Center procedures.
- Simulator data will be verified against actual plant startup data within 3 months of completing the Power Uprate Ascension Test Program.
- The applicable training programs and the simulator will be modified, or appropriate compensatory actions will be taken, in accordance with the

applicable Monticello Training Center procedures to reflect issues and discrepancies identified during startup testing within 6 months of completing the Power Uprate Ascension Test Program.

On the basis of the information discussed, the staff finds that the licensee has proposed satisfactory changes to the operator training program and the plant simulator as a result of the power uprate.

The staff concludes that the previously discussed review topics associated with the proposed Monticello plant power uprate have been or will be satisfactorily addressed. The staff further concludes that the power uprate should not adversely affect operator performance or operator reliability.

3.0 EVALUATION OF TRANSIENT AND ACCIDENT ANALYSES

The staff requested that the licensee identify all codes/methodologies used to obtain safety limits and operating limits and how it verified that these limits were correct for the appropriate uprated core. The licensee was also requested to identify and discuss any limitations associated with these codes/methodologies that may have been imposed by the staff. In a letter dated March 6, 1998 (Ref. 4), the licensee responded to this staff request. The licensee stated that it was the licensee's practice to review vendor calculations by surveillance audits and third-party technical reviews to ensure that the calculations are based upon approved methodologies and are consistent with all constraints. Restrictions and conditions applicable to GE's core and fuel design are documented in GESTAR II, NEDE-24011-P-A, Revision 11, General Electric Standard Application for Reactor Fuel, November 1995. Restrictions and conditions applicable to the SAFER/GESTR-LOCA methodology are described in Section 3.2 of NEDC-32514P Rev. 1. This is acceptable to the staff.

3.1 Reactor Transients

The limiting USAR transients were reevaluated using the staff-approved GEMINI transient analysis methods with uprated power input parameters. The transients were analyzed at the uprated power and maximum allowed core flow point on the power/flow operating map for uprated operational conditions. Table E-1 of NEDC-32424P (Ref. 16) provides the specific events to be evaluated for power uprate, the power level to be assumed, and the computer models to be used. The events identified in Reference 16 and the events evaluated in the USAR were reviewed for MNGP. The limiting events identified in Reference 16 were determined to be limiting for MNGP. Additional events were reviewed and confirmatory analyses performed to show that power uprate resulted in no new limiting events. The transients reviewed and analyzed for power uprate are listed in Table 9-1 of NEDC-32546P (Ref. 21). The input conditions for the transient analyses are summarized in Table 9-2 of NEDC-32546P, and are compared with the most recent reload fuel cycle analyses. Most of the transients were analyzed at the full uprated power and maximum allowed core flow operating point on the power/flow map. Direct or statistical allowance for 2-percent power uncertainty was included in the analysis. The SLMCPR in Table 9-2 was used to calculate the OLMCPR provided for the analyzed events. For all events three SRVs are assumed to be out of service. The effect of SLMCPR is generically evaluated in Section 3.4 of Reference 18 (NEDC-32523P).

The limiting events for each limiting transient category were analyzed to determine their sensitivity to core flow, feedwater temperature, and cycle exposure. The Reload 18 core was used as the representative core for power uprate. The limiting transient analysis results for the 1775 MWt power uprate are provided in Table 9-3 and Figures 9-1 through 9-3 of Reference 21. The limiting transient, feedwater controller failure-maximum demand, yielded the greatest change in critical power ratio (CPR). This change in CPR is added to the SLMCPR to provide the OLMCPR. Based on the above, the analyzed reactor transient responses at the power uprate conditions are acceptable.

3.2 Anticipated Transients Without Scram

A generic evaluation for the ATWS event is presented in Section 3.7 of NEDC-32523P (Ref. 18) for a BWR/3, similar to MNGP. This evaluation takes into account the ATWS mitigation features dictated by the ATWS rule of 10 CFR 50.62. The generic evaluation concludes that the ATWS acceptance criteria for fuel, RPV, and containment integrity will be met for power uprate if (1) reactor power increases are less than or equal to 20 percent, (2) dome pressure increase is less than or equal to 70 psi, (3) the SRV opening set points increase less than or equal to 105 psi, and (4) the ATWS high pressure set point increases less than or equal to 85 psi. The MNGP power uprate to 1775 MWt represents a power increase of 6.3 percent and requires no increase in dome pressure, nominal SRV opening set points, or ATWS high pressure set point. The plant parameter changes for power uprate are within the above criteria. Therefore, the analyzed response to an ATWS at the power uprate conditions is acceptable.

3.3 Station Blackout

The MNGP compliance method as approved by the NRC for the requirements delineated in 10 CFR 50.63 was considered in the evaluation of the station blackout (SBO) event. Plant response and coping capabilities for an SBO are affected by reactor operation at the bounding 1880 MWt due to an increase in decay heat. There are no changes to the systems and equipment used to respond to an SBO, nor is the required coping time changed.

The following areas contain equipment necessary to mitigate the SBO event: control room; cable spreading room; HPCI equipment room; steam tunnel; drywell and suppression pool; battery rooms, inverter areas. The temperatures in the control room and cable spreading room are not affected by a reactor power increase to 1880 MWt. The HPCI room, steam tunnel, and drywell area temperatures increase by a small amount. The peak suppression pool temperature increases; however, the systems used to respond after power is restored are designed for the power uprate suppression pool temperature. The reactor water makeup inventory requirement increases with the increase being within the available inventory. The licensee stated that the SBO analyses for power uprate show that all applicable core cooling and containment integrity criteria delineated by 10 CFR 50.63 continue to be met and that the plant response and coping capabilities for an SBO event are acceptable for a power increase to 1880 MWt. Therefore, the power uprate to 1775 MWt has no significant effect.

3.4 Adequate Core Cooling for Transients with a Single Failure

During a loss of feedwater (LOFW) event and assuming an additional single failure (loss of RCIC or HPCI), reactor water level is maintained above the top-of-the-active-fuel by automatic initiation of the RCIC or HPCI system. Because of the increased decay heat from power uprate, more time will be required for the automatic systems to restore water level. Once water level is restored, operator action is needed for long-term plant shutdown (control RPV water level, reduce RPV pressure, and initiate RHR shutdown cooling). These sequences of events do not require any new operator actions or shorter operator response times. Therefore, the operator actions for a LOFW transient do not significantly change for power uprate. The LOFW transient remains the bounding worst-case transient with single failure.

3.5 Radiological Analyses for Design-Basis Accidents

In Revision 1 to the license amendment request (Ref. 21) the licensee reevaluated the radiological consequences of the following four postulated design-basis accidents (DBAs) at an uprated reactor power level of 1918 MWt (102 percent of bounding reactor power of 1880 MWt) except the main steam line break accident (MSLBA). The MSLBA is analyzed at hot standby conditions since the postulated MSLBA from the hot standby condition results in a high rate of depressurization and rapid rising of water level to the main steam line inlet; therefore, maximum coolant mass is released through the break. The analyzed DBAs are (1) LOCA, (2) fuel handling accident, (3) control rod drop accident, and (4) MSLBA outside drywell. The licensee provided additional information on radiological consequence analyses in a separate submittal (Ref. 4) in response to the staff's request for additional information dated February 11, 1998.

In Exhibit A, Section III of Reference 21, the licensee concludes that the radiological consequences of an accident subsequent to implementation of MNGP power uprate are slightly increased (approximately proportional to the increase in reactor power) and that these consequences remain well below the dose criteria specified in 10 CFR Part 100 and GDC 19 of Appendix A to 10 CFR Part 50.

The staff has reviewed the radiological consequence analyses performed by the licensee and finds that the calculational methods used for the radiological consequence assessments are acceptable except for the licensee's assumption of nonorganic iodine removal by suppression pool water scrubbing. The licensee claimed a decontamination factor of 4.3 for nonorganic iodines for scrubbing and retention by suppression pool for the MSIV leakage pathway. No credit for leakage of fission products from the drywell to the suppression pool region can be assumed for the MSIV leakage pathway since it is bypassing the drywell (MNGP is designed with General Electric Mark I Containment). By letter dated July 30, 1998 (Ref. 15), the licensee committed to incorporate this correction when updating the Monticello USAR.

To verify the licensee's conclusions, the staff performed an independent radiological consequence analyses for the same four DBAs that the licensee analyzed at a reactor power level of 1918 MWt (MSLBA at hot standby conditions) in accordance with Standard Review Plan (SRP) (NUREG-0800) and using the staff positions in applicable current regulatory guides.

In its evaluation of the radiological consequences due to the MSIV leakage following a postulated LOCA, the staff allowed a credit for iodine holdup for decay and iodine deposition for plate-out in the main steam lines, the steam drain lines, and main condensers. This is a deviation from the SRP. The licensee also claimed a similar credit in its analyses using the methodologies and models developed by GE.

Section III(c) and VI of Appendix A to 10 CFR Part 100 requires that structures, systems, and components necessary to ensure the capability to mitigate the radiological consequences of accidents that could result in exposures comparable to the dose guideline exposures of Part 100 be designed to remain functional during and after an SSE. Thus, the main steam line, portions of its associated piping, and the main condenser are required to remain functional if the SSE occurs. Consequently, these components have been evaluated as described in Section 4.0 to assure that they would retain sufficient structural integrity following a safe shutdown earthquake to transport main steam isolation valve leakage to the condenser. In addition, Appendix A to 10 CFR Part 100 requires that the engineering method used to ensure that the safety functions are maintained during and after occurrence of an SSE involve the use of either a suitable dynamic analysis or a suitable qualification test.

For the purpose of providing a credit for iodine holdup and plate-out, the staff requires that the main steam piping (including its associated piping to the condenser) and the condenser remain structurally intact following an SSE, so they can act as a holdup volume for fission products.

The licensee provided additional information regarding the seismic verification of the MSIV leakage path in a separate submittal (Ref. 12) in response to the staff's request. The licensee concluded that the MNGP design provides reasonable assurance that the main steam piping from the outboard isolation valve up to the turbine stop valve, the main steam drain lines up to the condenser, and the main condenser will remain structurally intact; therefore, they can act as a holdup volume for fission products during and following an SSE. The staff's review of this area is documented in Section 4.0 of this safety evaluation.

The licensee submitted the site meteorological data and calculated atmospheric dispersion factors (X/Q values) (Ref. 30). The licensee stated that these meteorological data, analysis, and X/Q values are also applicable to the power uprate radiological consequence analysis. In the submittal, the licensee stated that it has used the methodology described in Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," for determining the site boundary and low population zone X/Q values and for calculating the X/Q values for the release from the offgas stack to the control room intake. The licensee used the methodology described in NUREG/CR-5055 (Ref. 28) for calculating the ground level release control room intake X/Q values. The staff has not accepted the methodology in NUREG/CR-5055 that has been revised into the ARCON96 methodology described in NUREG/CR-6331, Rev. 1 (Ref. 29).

The staff independently calculated X/Q values for the site boundary and low population zone using the methodology described in Regulatory Guide 1.145 and for the control room air intake using the ARCON96 methodology. The staff has found the licensee's X/Q calculations for the offgas stack and turbine building releases to be adequately conservative for this assessment. For the postulated turbine building release, fission products are conservatively assumed to be released at a point located in the center of four sealed off roof exhauster openings on the turbine building roof closest to the control room air intake.

The staff finds that the differences in the control room X/Q values calculated by the licensee and staff are within the uncertainty ranges of mixing of fission products with air in the turbine building prior to release to the environment. The staff has not provided and the licensee has not claimed any mixing credit in the turbine building. The staff used the X/Q values calculated by the licensee in the staff dose assessment for this power uprate analysis. The resulting radiological consequence analyses are provided in Table 3.5-1 and the major parameters and assumptions used by the staff are provided in Table 3.5-2 through Table 3.5-6.

The staff concludes that the proposed power uprate at MNGP will provide reasonable assurance that the radiological consequences of bounding DBAs will not exceed dose acceptance criteria specified in the SRP, 10 CFR Part 100, and GDC 19 of Appendix A to 10 CFR Part 50. This conclusion is based on the staff's review of the radiological consequence analyses submitted by the licensee and the staff's independent confirmatory analyses. Therefore, the staff finds that the proposed power uprate is acceptable.

Tables 3.5-1 through 3.6-6 follow.

Table 3.5-1 Radiological Consequences of Design-Basis Accidents
(rem)

Postulated Accidents	EAB		LPZ		Control Room	
	Thyroid	WB	Thyroid	WB	Thyroid	WB
Loss of Coolant	3	< 1	29	< 1	26	< 1
Control Rod Drop	< 1	< 1	< 1	< 1	1.2	< 1
Main Steam Line Break	27	< 1	2.3	< 1	11	< 1
Fuel Handling	1.4	< 1	< 1	< 1	< 1	< 1

Table 3.5-2 Assumptions used in computing main steam line break accident

Parameter	Value
Power level, MWt	66.8 at hot standby
MSIV closure time, sec	10.5
Reactor primary coolant iodine concentrations (μCi/gm DEI-131)	2.0
Total mass release, lbm	8.62E+4
Total steam release, lbm	1.46E+4
Total liquid release, lbm	7.16E+4
Iodine partition factor	1.0
Dose conversion factor	FGR 11 and 12
Breathing rate, m^3/sec	3.74E-4
Atmospheric dispersion values	
0 to 2 hours, sec/m^3 EAB	9.20E-4
0 to 2 hours, sec/m^3 LPZ	7.93E-5

Table 3.5-3 Assumptions used in computing rod drop accident doses

Parameter	Value
Power level, MWt	1918
Peaking factor	1.5
Number of fuel rods failed	841
Number of fuel rods melted	9
Fraction of fission-product inventory released to coolant from perforated fuel rods	
Iodines, percent	10
Noble gases, percent	10
Fraction of fission-product inventory released to coolant from melted fuel rods	
Iodines, percent	50
Noble gases, percent	100
Initial reactor coolant iodine activity, μCi/gm (DEI-131)	2
Coolant iodine fraction reaching condenser (percent)	10
Iodine partition factor in condenser	0.1
Condenser leak rate, percent/day	1.0
Atmospheric dispersion values	
0 to 2 hours, sec/m^3 EAB	9.20E-4
0 to 2 hours, sec/m^3 LPZ	7.93E-5
2 to 8 hours, sec/m^3 LPZ	5.35E-5
1 to 4 days, sec/m^3 LPZ	2.28E-5
4 to 30 days, sec/m^3 LPZ	6.68E-6

Table 3.5-4 Assumptions used to evaluate the loss-of-coolant accident

Parameter	Value
Power level, MWt	1918
Fraction of core inventory released, fractions	
Noble gases	1.0
Iodine	0.25
Iodine chemical forms, fractions	
Organic	0.04
Elemental	0.91
Particulate	0.05
Primary containment leakage, percent/day	1.2
Primary containment free volume, ft^3	1.34E+5
Pressure suppression chamber, ft^3	1.03E+5
Suppression pool water volume, ft^3	6.80E+4
Suppression pool decontamination factors	
Containment leak pathway	4.3
MSIV leak pathway	1
ECCS leak rate, gpm	1.0
Iodine partition factor for ECCS leakage	0.01
SGTS flow rate, cfm	3.5E+3
SGTS filter efficiencies, percent	
Organic	85
Elemental	85
Particulate	85

Table 3.5-4 Assumptions used to evaluate the loss-of-coolant accident (continued)

Parameters	Values
Total MSIV leak rate, scfh	46
Condenser effective volume, ft^3	7.7E+4
Condenser leak rate, percent/day	2.99
Accident duration, days	30
Atmospheric dispersion values	
Ground level releases for MSIV leakage pathway	
0-02 hour EAB, sec/m^3	9.20E-4
0-08 hour LPZ, sec/m^3	7.93E-5
8-24 hour LPZ, sec/m^3	5.35E-5
1-04 day LPZ, sec/m^3	2.28E-5
4-30 day LPZ, sec/m^3	6.68E-6
Stack releases for containment and ECCS leakage pathways	
0-0.5 hour EAB, sec/m^3	1.01E-4
0.5-2 hour EAB, sec/m^3	3.21E-6
0-0.5 hour LPZ, sec/m^3	3.51E-4
0.5-8 hour LPZ, sec/m^3	1.30E-6
8-24 hour LPZ, sec/m^3	8.54E-7
1-04 day LPZ, sec/m^3	3.70E-7
4-30 day LPZ, sec/m^3	1.11E-7

Table 3.5-5 Assumptions used in computing fuel handling accident doses

Parameter	Value
Power level, MWt	1918
Peaking factor	1.5
Number of fuel rods damaged	125
Reactor shutdown time before fuel movement (hours)	24
Core percent released from damaged rods	
Iodine	10
Noble gases	10
Pool decontamination factor	
Elemental and particulate iodines	100
Organic iodine	1
Noble gases	1
Duration of accident, hours	2
Atmospheric dispersion values	
0-0.5 hour EAB, sec/m^3	1.01E-4
0.5-2 hour EAB, sec/m^3	3.21E-6
0-0.5 hour LPZ, sec/m^3	3.51E-4
0.5-2 hour LPZ, sec/m^3	1.30E-6

Table 3.5-6 Assumptions used to estimate the radiological consequences to control room operators

Parameters	Values
Control room free volume	2.7E+4 ft^3
Air intake flow	900 cfm
Filter efficiencies, percent	
Organic	98
Elemental	98
Particulate	98
Recirculation flow	0
Breathing rate of operators in control room for the course of the accident	3.47E-4 m^3/sec
Unfiltered infiltration rate	
0 to 8 hours	250 cfm
8 to 720 hours	10 cfm
Control room operator occupational factors	
0 to 24 hr	1
24 to 96 hr	0.6
96 to 720 hr	0.4
Atmospheric dispersion values, sec/m^3	
Ground level releases	
0-08 hour	1.67E-3
8-24 hour	1.41E-3
1-04 day	9.65E-4
4-30 day	5.62E-4
Stack releases	
0-0.5 hour	2.98E-4
0.5-8 hour	2.47E-11
8-24 hour	1.55E-11
1-04 day	6.20E-12
4-30 day	1.66E-12

4.0 SEISMIC VERIFICATION OF THE MSIV LEAKAGE PATH

The Monticello power uprate radiological analysis takes credit for deposition and holdup of radioactive iodine in the steam lines downstream of the main steam isolation valves (MSIVs) and in the main condenser. The main condenser and the pathway from the MSIVs were evaluated to assure that they would retain sufficient structural integrity following a safe shutdown earthquake (SSE) to transport the MSIV leakage to the condenser.

Because the original design basis of certain main steam system piping, equipment, and components that comprise the leakage pathway is not in accordance with Seismic Category I requirements, NSP has performed evaluations and seismic verification walkdowns to demonstrate that these main steam system piping, equipment, and components are seismically rugged.

The licensee used methodology suggested in the BWROG [Boiling Water Reactor Owners' Group] Report, NEDC-31858P, Rev. 2, entitled "BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems," (Ref. 31), to seismically evaluate this pathway. The licensee's submittal of June 15, 1998 (Ref. 12) discusses the applicability of this methodology for Monticello and summarizes the seismic evaluation that was performed for the piping systems and equipment in the MSIV leakage path for Monticello. The licensee stated that a reliable pressure boundary can be maintained in the pathway for the MSIV leakage to reach the condenser during and after an SSE seismic event.

The BWROG report has not been approved by the staff. However, based on a preliminary review to date, the staff has found the BWROG approach of utilizing the earthquake experience-based methodology to demonstrate the seismic ruggedness of non-seismically analyzed main steam system piping and main condensers, in addition to supplemental plant-specific seismic evaluations, to be acceptable for this amendment request.

The above methodology relies, in part, on the use of earthquake experience data and similarity principles. In addition, plant-specific analyses of piping and equipment was used in combination with the experience database method. Guidance on the use of experience database method for qualification of piping systems is described in Reference 31, and in the supporting documents cited therein. The Seismic Qualification Utility Group (SQUG) Generic Implementation Procedure (GIP) described in Reference 32 that was developed for the implementation of Unresolved Safety Issue (USI) A-46, was used to demonstrate the seismic ruggedness of certain existing equipment in the MSIV leakage path.

For Monticello, the primary components in the MSIV leakage path that are relied upon for pressure boundary integrity are the main condenser, main steam lines from the MSIVs to the turbine stop valves and to the turbine bypass valves, and the drain lines to the condenser. The drain lines originate from each of the four main steam lines. These drain lines are located downstream of the MSIVs and connect into a drain header that connects to the condenser. The leakage path utilizes three separate drain lines from the main steam piping to the drain header. These three drain lines include the main steam drain lines, the main steam cross tie drain, and the turbine bypass line drain. Each of these lines can be isolated by motor-operated valves (MOVs). Each MOV has a bypass line with a restricting orifice. Since the MOVs are not powered by essential power and are normally closed, it is assumed that the leakage will be through the MOV bypass lines via the restricting orifices. This provides a passive pathway for the MSIV leakage to reach the condenser because no valve positioning or operator action is necessary to establish the pathway. Therefore, periodic testing to demonstrate valve operability is not required.

The branch lines which interconnect with the MSIV leakage path are included in the scope of the system piping that is reviewed. These branch lines include the connection from the pathway to locations such as a closed valve that would assure that the MSIV leakage would be confined within the branch lines, and leakage would be transferred to the condenser.

The turbine bypass valves are normally closed and fail closed. Because these types of valves are not well represented in the experience data, the licensee conservatively assumed that the valves would fail open as a result of a postulated seismic event, and leakage would therefore go past the turbine bypass valves to the condenser. This portion of the piping was, therefore, also included in the evaluation.

4.1 Earthquake Ground Motion

This section of the safety evaluation contains a review of the earthquake data to assure that the vibratory ground motion, experienced at each of the facilities with equipment being used as a surrogate for similar equipment at Monticello, did indeed exceed the Monticello SSE. The ideal case, for this type of comparison, is to have actual recordings of the earthquake ground motion made at each of the facilities. The licensee has indicated that it relied on the ground motion estimates in the data base from actual instrument recordings at or near five facility sites and the SQUG Bounding Spectrum from the GIP-2 (Ref. 32) to verify the adequacy of the MSIV leakage path equipment.

The ground motion from an earthquake at a particular site is a function of the earthquake source characteristics such as the magnitude, focal mechanism, radiation pattern, stress drop, location of asperities and fault rupture history, and depth and orientation of the fault. It is also a function of the distance of the facility to the fault and the propagation properties of the rocks between them. The geology immediately under the facility site can also have a large effect on the amplitude and frequency content of the ground motion. Two of the more appropriate methods of estimating earthquake ground motion where there are no nearby recordings involve the use of (1) calibrated numerical modeling of the fault rupture and wave propagation process, and (2) empirical attenuation relationships obtained from the statistical analysis of large sets of earthquake data.

The licensee has stated that the Monticello condenser design is similar to, or bounded by, data for Moss Landing Units 6 and 7 which experienced the Loma Prieta 1989 earthquake, and Ormond Beach Units 1 and 2 which experienced the Point Mugu 1973 earthquake. They also indicate that the earthquake experience data that is directly being used for comparison to the Monticello piping is obtained from the following site-earthquake pairs.

El Centro Steam Plant - Imperial Valley 1979 earthquake.
Valley Steam Plant - San Fernando 1971 earthquake.

Moss Landing Power Plant - Loma Prieta 1989 earthquake.
Ormond Beach Power Plant - Point Mugu 1973 earthquake.
Humboldt Bay Power Plant - Ferndale 1975 earthquake

For other equipment included in the scope of the leakage path review, the Bounding Spectrum in the SQUG GIP-2 was used to verify their seismic adequacy.

The ground motion estimate at the El Centro Steam Plant from the Imperial Valley 1979 earthquake was based on a recording made at a U. S. Geological Survey (USGS) strong ground motion station about 1 kilometer from the facility. Because of the density of seismic recordings in that area and the distribution of the ground motion the staff concludes that the estimate for the site is significantly larger than the Monticello SSE and is appropriate for use in verifying the seismic adequacy of the Monticello equipment, in the MSIV leakage pathway, similar to that in the El Centro Steam Plant.

The ground motion estimate developed by EQE [Earthquake Engineering], Inc. at the Valley Steam Plant from the San Fernando 1971 earthquake, for use in the USI A-46 program, was based on an extrapolation of data from a relatively distant location. In 1988, the USGS performed studies to estimate the ground motion at selected sites from the San Fernando 1971 earthquake in support of the NRC's resolution of USI A-46. The USGS estimate of the ground motion at the Valley Steam Plant is lower than the EQE, Inc. estimate. As can be seen in Figure 3-1 of Ref. 12, the USGS estimate is significantly higher than the Monticello SSE spectrum. As in the resolution of the A-46 ground motion issue, the staff still considers the USGS estimate to be the characterization of the ground motion at the Valley Steam Plant from the San Fernando 1971 earthquake and that the estimate is appropriate for use in verifying the seismic adequacy of the Monticello equipment, in the MSIV leakage pathway, similar to that in the Valley Steam Plant.

The ground motion estimate at the Moss Landing Steam Plant from the Loma Prieta 1989 earthquake is based on a study performed by Pacific Gas and Electric Company (PG&E) (Ref. 33), the owner of the Moss Landing Steam Plant at the time of the earthquake. A copy of the report of the study was provided by PG&E for the NRC staff's use. The analysis performed by PG&E was technically sound and comprehensive. It shows a thorough understanding of the problem and the staff concludes that their estimate of the ground motion is appropriate for use in verifying the seismic adequacy of the Monticello equipment, in the MSIV leakage pathway, similar to that in the Moss Landing Steam Plant.

The NRC has copies of the strong ground motion recordings and response spectra at Humboldt Bay from the Ferndale 1975 earthquake. The ground elevation vibratory motions experienced at the plant were all larger than the Monticello SSE ground motions. Therefore, the ground motion at the Humboldt Bay Plant from the Ferndale 1975 earthquake is appropriate for use in verifying the seismic adequacy of the Monticello equipment, in the MSIV leakage pathway, similar to that in the Humboldt Bay Plant.

The NRC does not have recordings or response spectra from the Point Mugu 1973 earthquake at the Ormond Beach Plant. To make an estimate of the level of the vibratory motion at the Ormond Beach Plant, the staff used information in its possession about the source characteristics of the earthquake and ground motion attenuation relations derived from western United States empirical earthquake data. The response spectra estimated by the staff are significantly higher in amplitude than the Monticello SSE design spectrum and, therefore, can be used to verify the seismic adequacy of the Monticello equipment, in the MSIV leakage Pathway, similar to that in the Ormond Beach Plant.

In 1988 as part of the staff review of the SQUG earthquake data base for use in the resolution of USI A-46, the NRC asked the USGS to make independent ground motion estimates for the data base earthquake facility pairs. In general, depending on the site and the frequency, the USGS estimates exceeded or were less than the SQUG estimates. The average of the USGS estimated ground motion was compared to the GIP-2 Reference Spectrum and was found to exceed it at all frequencies. On this basis, the NRC concluded that the GIP-2 Reference Spectrum and the GIP-2 Bounding Spectrum (lower bound of the SQUG data base spectra) were acceptable for verifying the seismic adequacy of the equipment in the SQUG data base facilities. As can be seen in Figure 3-2 of Reference 12, the GIP-2 Bounding Spectrum is significantly higher than the Monticello SSE spectrum. The staff, therefore, considers it appropriate to use the GIP-2 Bounding Spectrum in verifying the seismic adequacy of the Monticello equipment, in the MSIV leakage pathway, similar to that in the SQUG GIP-2 data base.

Based on our independent analysis of the earthquake experience database, the staff concludes that Monticello SSE demand is well below the seismic ground motion which was experienced at the facilities discussed above. Consequently, the use of the database as proposed by the licensee to verify the seismic adequacy of the equipment and piping in the MSIV leakage pathway is acceptable for this Monticello license amendment.

4.2 Seismic Verification Walkdowns

In order to confirm the functional capability of the leakage pathway, the licensee performed a seismic verification walkdown of the MSIV leakage pathway. The licensee indicated in its submittal (Ref. 12) that the walkdown was required for the evaluation of the seismic capacity of the subject piping and components. To assess the seismic capacity of the system, the walkdown focused on verifying that the pathway piping and components fall within the bounds of the design characteristics of the seismic experience database as discussed in Section 6.7 of the BWROG Report (Ref. 31). Specifically, the walkdown was performed to (1) verify that Monticello piping systems have attributes similar to those in the earthquake experience databases that have demonstrated good seismic performance, (2) verify the general conformance of the piping system spans to the requirements of ANSI B31.1, and (3) examine the leakage system from the outboard MSIVs to the condenser to identify potential seismic vulnerabilities, considering those structural details and causal factors that resulted in component damage at database plants.

The licensee indicated that worst case piping and component supports were identified. Specifically, rod hangers susceptible to fatigue failure, "hard spot" short rod hangers, and U-bolts subjected to significant lateral loads were identified for further, detailed evaluations. The potential vulnerabilities identified by the licensee, as a result of walkdowns, also included categories such as support failure, inadequate support, corrosion/erosion of piping, and spatial interaction, etc. Items identified as outliers, and their resolutions are presented in Table 5-8 of Reference 12.

The staff found the walkdowns performed by the licensee, as well as the corrective actions taken for the identified outliers, including the detailed analytical evaluations performed for the outliers, to be acceptable.

4.3 Seismic Demand

All items in the leakage pathway were evaluated for the SSE demand. The SSE ground response spectrum is identified in the Monticello Updated Safety Analysis Report. The Monticello SSE horizontal peak ground acceleration (PGA) is 0.12g. The vertical peak ground acceleration was taken as 2/3 of the horizontal demand.

4.3.1 Piping and Support Seismic Demand

The majority of the Monticello MSIV leakage path piping is located in the turbine building, recombiner building or buried. A small amount of the piping is located in the reactor building including the steam tunnel. Different seismic demands used for the evaluation of piping and supports are presented separately in Reference 12. The licensee indicated that the seismic adequacy of these components was established as a result of the following evaluations:

(a) Limited Analytical Reviews of Piping

In performing the limited analytical reviews of piping in the turbine building and the recombiner building (all of which are located less than 40 ft above grade) using a dynamic analysis technique, the horizontal piping demand is based on the 5-percent damped Monticello ground response spectrum multiplied by a factor of 1.5. This method of estimating median-centered amplified floor spectra was used because amplified floor response spectra for these buildings at Monticello do not exist. The vertical demand is 2/3 of the horizontal demand.

When the static analysis technique was employed in performing the limited analytical reviews of piping systems, the demand static load coefficient in the horizontal direction was 1.5 times the maximum spectral acceleration of the ground response spectrum. In the vertical direction, 1.5 x 2/3 of the maximum spectral acceleration of the ground response spectrum was used.

For piping in the reactor building, the horizontal demand was based on the applicable 5-percent damped floor response spectrum (FRS), and the vertical demand was 2/3 of the horizontal demand.

The staff found the licensee's definition of seismic demands for various piping runs in the MSIV leakage pathway to be acceptable.

(b) Limited Analytical Review of Buried Piping System

In the licensee's evaluation of buried piping systems, the seismic demand is the design-basis SSE ground response spectrum. This is acceptable to the staff.

(c) Worst-case Support Reviews

In determining the horizontal seismic loads for the worst case support reviews, the total weight per unit length of piping including material weight, fluid weight, insulation weight, and any other weights in the piping system was multiplied by the maximum spectral acceleration of an applicable horizontal response spectrum. The applicable horizontal spectrum for all piping except that in the reactor building was 1.5 times the 5-percent damped ground response spectrum. For piping supports in the reactor building, the applicable amplified FRS was used. For vertical loads, 2/3 of the horizontal value was used. This is acceptable to the staff.

4.3.2 Condenser Demand Spectra

The Monticello condenser is located below grade at the lowest level of the turbine building (Elevation 911 feet above the sea level). The licensee's applied seismic demand is the Monticello SSE ground spectrum, and is acceptable to the staff.

4.3.3 Related Equipment Demand Spectra

Applied seismic demand for related equipment is based on the SSE ground spectrum and the corresponding FRS. Consistent with the Monticello USAR, the reactor building FRS at an equivalent elevation is used by the licensee to define the FRS for equipment in the turbine and recombiner buildings. This FRS was also used for the Monticello USI A-46 resolution (Ref. 34). In addition and consistent with the GIP-2 methodology, 1.5 times the ground spectrum was optionally used as "realistic, median centered" demand for some equipment items meeting the GIP-2 below 40-foot-above-grade elevation limitation and the 8 Hertz lower bound frequency limitation. This was only done for equipment at or below grade. The licensee stated that, as with the piping, the largest majority of the equipment is located at the lowest elevations in the buildings.

The turbine building below the 951 ft elevation is a stiff reinforced concrete shear wall structure typical of nuclear plant structures. All related equipment in the turbine building is located at or below the 951 ft elevation. The recombiner building is also a stiff reinforced concrete shear wall building that is typical of nuclear plant structures. On this basis, the staff found the use of the above (1.5) factor acceptable.

4.4 Seismic Capacity

To ensure that the seismic demands for the Monticello leakage path piping systems and equipment are met, their corresponding experience-based capacities were established. The design criteria for the evaluations of piping systems and supports, however, are discussed in Section 4.6, Analytical Evaluations.

4.4.1 Building Seismic Capacity

The seismic integrity of the buildings that house the leakage pathway piping systems and equipment was first evaluated. In Reference 12, it is stated that the MSIV equipment and piping are confined to three buildings: the reactor, turbine, and recombiner buildings. The reactor and recombiner buildings were designed as seismic Category I structures. The turbine building consists of two parts: concrete and steel. The turbine building was built with reinforced concrete from the ground up to the operating floor and this portion of the building was considered as a seismic Category I structure because portions of the building, such as the switchgear room, were designed as seismic Category I structures, and are within the concrete portion of the building. The remaining portion of the turbine building above the operating floor was constructed with structural steel. This portion of the structure was seismically evaluated for the SSE accelerations of the reactor building at equivalent elevations. The licensee stated that all of the piping and equipment relevant to the MSIV leakage path are located within the concrete portion of the turbine building. Only a few instrument lines are located at the operating floor of the turbine building. Based on the information provided to the staff as described above, the staff concludes that these buildings are adequate for this MSIV amendment request.

4.4.2 Experience-Based Piping Capacity

In its submittal (Ref. 12), the licensee asserted that experience from past strong earthquakes at conventional power plant and industrial facilities indicate those piping systems designed to industrial standards are rugged. The licensee stated that this experience data includes piping systems which were not specifically designed for seismic loads. In all strong motion earthquakes affecting power stations in the United States since 1952, the amount of piping system failures observed was a very small percentage of the total piping at risk (Ref. 31).

For the MSIV leakage pathway, the licensee conducted walkdowns to compare the subject piping systems to piping systems which have actually experienced strong motion earthquakes and to verify the seismic adequacy of the main steam leakage path piping. This method utilizes a capacity vs. demand spectrum comparison, augmented by extensive walkdowns, worst-case calculations, and documentation to assure acceptable piping spans, piping support configurations, design attributes, and the absence of known seismic vulnerabilities. Section 4.3.1 of this evaluation discusses the capacity spectra that were used in the establishment of the piping seismic capacity.

4.4.3 Experience-Based Condenser Capacity

An evaluation of the seismic ruggedness of condensers and condenser anchorage for GE BWR plants is reported in Reference 31. The configurations of the GE BWR condensers were compared to condensers in the earthquake experience data. Condensers in the earthquake experience data exhibited substantial seismic ruggedness even when they were not designed to resist earthquakes. Comparisons of condenser designs in GE BWR plants with those in the earthquake experience database revealed that GE plant designs are similar to those that exhibited good earthquake performance. The study concluded that a failure and significant breach of a pressure boundary in the event of a design-basis SSE is highly unlikely. This conclusion was further verified by a detailed comparison of the Monticello condenser configuration to the earthquake experience data. A detailed evaluation of the Monticello condenser anchorage capacity was also performed by the licensee (see Section 4.7.2). Based on the above, the staff concludes there is reasonable assurance that the condenser is seismically adequate for the proposed MSIV leakage path system.

4.4.4 Experience-Based Capacity of Related Equipment

Other equipment in the scope of the leakage pathway review included valves, instruments, and tanks which are referred to as related equipment in this evaluation. The SQUG GIP-2 methodology, documented in Reference 32, was employed to address the seismic adequacy of the equipment. GIP-2 provides a formal procedure for evaluating these classes of equipment against the earthquake experience data. The licensee's implementation of the GIP-2 procedure at Monticello is separately documented in Reference 34.

In Reference 12, the licensee compared the Monticello SSE ground spectrum to the GIP-2 Reference Spectrum and GIP-2 Bounding Spectrum, and found that the Monticello spectrum is well bounded by the GIP-2 spectra. The staff finds this acceptable.

4.5 Comparison of Monticello and Experience Data

In Reference 12, the licensee provided a database for the main steam and process piping at the above-mentioned database plants, i.e., Valley Steam Plant, Ormond Beach Power Plant, El Centro Steam Plant, Moss Landing Power Plant, and Humboldt Bay Power Plant. Table 5-1 of Reference 12 presents a summary of the various piping, sizes, schedules and diameter-to-thickness (D/t) ratios for each of the walkdown packages. Table 5-2 presents a general summary of the same data for the piping systems which constitute the experience data. In Table 5-3, the licensee also presents a comparison of the D/t ranges of the Monticello piping to the experience data piping. The Monticello piping systems in the leakage path are enveloped by the experience data with the following exceptions.

(1) The experience data does not specifically identify the existence of 3-1/2 in. and 5 in. diameter piping.

(2) The Monticello 1 in. piping has a lower bound D/t ratio of 4 versus 5 in the experience data.

(3) The Monticello 24 in. piping has a lower bound D/t ratio of 20 versus 23 in the experience data.

(4) The Monticello 18 in. piping has an upper bound D/t ratio of 48 versus 43 in the experience data.

The licensee stated that for items (2) and (3), these lower D/t ratios are due to the use of thicker wall piping which would be stronger and have higher capacity than the experience data piping and therefore is not a concern. For item (4), the exceedance is only 12 percent which is less than typical piping system fabrication tolerances. Therefore, the licensee considers that this piping is adequately represented in the experience data. The 3-1/2 in. and 5 in. diameter piping, are not explicitly represented in the database; however, their D/t ratios are enveloped by those of larger and smaller sizes of piping. In addition, the 3-1/2 in. and 5 in. piping is in the steam seal system that was analyzed in detail. Therefore, the staff considers this piping to be adequately enveloped by the experience data and the supporting analysis.

Table 5-4(a) of Reference 12 provides a summary of the allowable stress capacity of the predominant piping materials of the experience data piping. Table 5-4(b) provides a similar summary for the Monticello piping. These tables demonstrate that the Monticello piping in the leakage pathway is adequately represented in the piping experience data.

In addition, Table 5-5 of Reference 12 provides a summary of minimum and maximum ratios of the actual Monticello vertical support spans to the suggested ANSI B31.1 deadweight spans and the actual Monticello lateral support spans to the suggested ANSI B31.1 spans. Figures 5-1 through 5-4 compare the Monticello vertical support span ratios (VSR) and lateral-to-vertical support span ratios (LVSSR), for both small-bore and large-bore piping, corresponding to the experience piping support span ratio data. These figures show that the Monticello piping support spans are well represented and adequately enveloped by the piping experience data, and are acceptable to the staff.

The above experience span ratio data was also used as a "screening criteria" to identify systems or portions of systems whose seismic capacity may require additional review or limited analytical evaluation. Therefore, in the licensee's review of the Monticello piping systems, any welded steel piping system having a VSR and an LVSSR in excess of 1.5 and 6.0, respectively, was subjected to an in-depth review and evaluation. For a threaded steel piping system, the corresponding VSR and LVSSR values are 1.5 and 4.0, respectively. The screening criterion was also used to screen systems that required additional supports to ensure that they had adequate seismic capacity. The staff found the licensee's approach acceptable.

4.6 Analytical Evaluations

All piping systems in the MSIV leakage pathway were evaluated to the screening criteria discussed in Section 4.5 above, to determine whether a detailed analysis is required to establish the seismic ruggedness for piping configurations that were not found acceptable using the seismic experience approach.

4.6.1 In-depth Piping Analyses

With the exception of the main steam lines between the MSIVs and the main turbine which have been previously evaluated to meet the requirements of ASME Code Class 1 loading, including SSE loads, the majority of piping systems under review were originally designed to the 1967 B31.1 Power Piping Code. The original design only considered loadings due to pressure, dead load, design mechanical loads, and thermal loads. The capacity criteria used for piping system limited analytical reviews and detailed analyses, which included consideration of a design basis SSE, are provided in Reference 12. These criteria limit the resultant stresses due to pressure and deadweight and seismic inertia loads to approximately the yield stress, and assures elastic behavior of the piping system during and after a design basis SSE event. The criteria also limit the combined effects of primary and secondary stresses to approximately 1.2 times the yield stress, which effectively assures an elastic shakedown, and that no significant membrane stress rupture will occur.

Detailed response spectra modal analyses were performed for several piping systems. Table 5-9 of Reference 12 provides a summary of these analyses and the associated bases.

In addition, localized equivalent static analyses were used to (1) evaluate the effects of seismic anchor motions (SAMs), (2) evaluate spatial interaction conditions, (3) evaluate localized areas of seismic vulnerability, and (4) determine loads used in the detailed support evaluations. Table 5-10 provides a summary of the equivalent static analyses performed. In addition, limited analytical reviews were performed for portions of other piping systems which involved complex spatial interactions, or for which a highly accurate prediction of piping support loads was required. The licensee stated in Reference 14 that the analytical results indicated that the piping systems presented in Table 5-9 all have adequate capacity to maintain leak-tight structural integrity during and after a postulated SSE.

The licensee indicated in Reference 12, that the steam seal discharge system piping was selected as the worst case piping system, based on the following considerations: (1) It is a large, complex piping system containing multiple pipe sizes, from 1 in. to 12 in.; (2) it is one of the most flexible systems reviewed and therefore has the potential for significant seismic response; (3) it contains a significant number of small bore branch lines, and significant seismic anchor motion stresses may be induced in the branch lines; (4) the potential exists for several spatial interactions between this system and other systems, structures or components; and (5) it is located at an upper elevation, and is expected to experience a greater level of input seismic excitation. The piping was analyzed using the response spectrum modal analysis technique. Table 5-12 provides a summary of maximum stresses determined from this analysis, which are shown to be within the allowable limits discussed above. The staff has reviewed the above information and found it to be acceptable.

In addition, the licensee performed one worst-case analytical review for all buried piping systems. A detailed enveloping dynamic analysis was performed for buried portions of the piping systems contained in six of the walkdown packages. These analyses include soil-structure interaction effects for the turbine, reactor, and recombiner buildings, and an evaluation of both the seismic wave passage effects and the effects of relative motions between the buildings from which the piping exited and entered. The results of the analysis demonstrate that the relative motions, resulting from a design SSE, between buildings at the buried depth of the piping were very small and do not induce significant stresses in the piping. The analyses also indicated that the maximum stresses in the piping due to wave passage effects would produce stresses in the piping on the order of 11,000 psi which are significantly below the capacity criteria presented in Reference 12. The staff found this to be acceptable.

4.6.2 Detailed Support Qualifications

In Reference 12, the licensee stated that selection of supports for detailed support qualifications were based on identifying or establishing worst case support during the walkdowns. The basis for the determination of these worst case supports included consideration of the following: (1) short, fixed, or hard spot rod hangers that were judged to be susceptible to fatigue failure during a design basis SSE event; (2) U-bolts susceptible to significant lateral loads; (3) supports that were judged to be the most susceptible to failure during a design basis seismic event based on field review; and (4) supports on piping systems for which detailed seismic analyses were performed.

For piping supports, the acceptance criteria used in the worst case support evaluation are based on the allowables of the AISC Steel Construction Manual, which will assure that the maximum stresses in the support members are at or slightly less than the material yield stress. This was found to be acceptable.

Table 5-11 of Reference 12 provides a summary of the number of supports subjected to detailed analytical reviews and the basis for their inclusion in the review. These supports represent approximately 15 percent of the support population in the MSIV leak path. All supports including the worst case supports given in Table 5-11 were evaluated. In Reference 14, the licensee stated that these supports were found to have met the support design criteria presented in Reference 12. Meeting these criteria assures that adequate safety factors exist. For modifications to piping systems described in Table 5-8 of Reference 12, the licensee also stated in Reference 14 that analytical evaluations were performed, and that all modified piping systems meet the support design criteria presented in Reference 13. In its letter dated July 30, 1998 (Ref. 15), the licensee stated that modifications described in Table 5-8 of Reference 12 have been completed. The staff found the licensee's approach for resolution to be acceptable.

4.7 Anchorage Evaluation

As stated in Reference 12 the licensee concluded that the concrete anchor bolts used for pipe supports were "Philips-Redheads" of the self drilling type. To provide a consistent factor of safety between the piping support anchorage and the equipment anchorage, the anchorage bolt capacities of Appendix C of SQUG-GIP-2 were used for both piping and equipment supports.

4.7.1 Anchorage for Piping Supports

In Reference 14, the licensee states that all pipe support anchorages were reviewed as part of the walkdown effort, and 130 anchorages, which were determined to be worst case samples, were selected for detailed evaluations. The licensee states that all pipe support anchorages meet the evaluation criteria. The evaluation criteria for welded and bolted anchorages were the same as those provided in the AISC Steel Construction Manual except the allowable stress was increased by a factor of 1.7 when an SSE load was included. The GIP-2 criteria for concrete expansion anchor bolts developed by SQUG were used for the evaluation of the adequacy of concrete expansion anchor bolts. These criteria are acceptable to the staff. Since the licensee has reviewed all the piping support anchorages and analyzed 130 worst-case sample anchorages against acceptable evaluation criteria, the staff concludes those piping support anchorages are adequate for this MSIV leakage path evaluation.

4.7.2 Anchorage for Condenser

The condenser consists of a high pressure and a low pressure shell. For each shell, the licensee calculated the forces required both in tension and in shear for the anchor bolts for the design loading condition of operating loads plus SSE (Ref. 14). The required forces are 112 kips in tension and 235 kips in shear for the high pressure shell and 143 kips in tension and 223 kips in shear for the low pressure shell. The licensee also calculated the capacities of the condenser anchor bolts. They are 245 kips in tension and 296 kips in shear. The licensee has demonstrated that the condenser anchorage capacities are greater than the corresponding design values. Therefore, the condenser anchorages are adequate for this MSIV leakage path evaluation.

4.8 Conclusion

Based on the above evaluation, the staff concludes that there is reasonable assurance that the Monticello main steam lines, main steam drain lines, condenser, and the associated interconnected piping and supports will be seismically adequate for the proposed MSIV leakage system. The staff's conclusion is based on (1) the staff's independent analysis of the earthquake experience database confirmed that the SSE demand at Monticello is well below the seismic ground motion that was experienced at the facilities in the earthquake experience database, (2) the design attributes of the Monticello main condenser are generally enveloped by those of the condensers in the earthquake experience database, and that the condenser assembly has sufficient anchorage capacity, (3) the nonseismically analyzed leakage path pipes are represented by those in the earthquake experience database that demonstrated good seismic performance, (4) the detailed analyses and the worst-case analysis performed for the nonseismic portion of the main steam drain lines indicated adequate safety margins for piping stresses and support loads, and (5) the turbine building has been adequately designed to withstand the SSE loads. The staff, therefore, concludes that the licensee's proposed MSIV leakage path system is acceptable.

It should be noted that the staff's acceptance of the experience-based and GIP-2 methodology as presented by the licensee in this amendment is restricted to its application for ensuring the pressure boundary integrity and functionality of the MSIV leakage path system. The staff's acceptance of the methodology for this application is not an endorsement for the use of the experience-based methodology for other applications at Monticello.

5.0 INDIVIDUAL PLANT EVALUATION

The licensee performed an evaluation to assess the potential effects on plant risk of an MNGP power uprate to 1775 MWt, which represents a 6.3-percent increase in reactor thermal power. This evaluation was performed at a bounding value of 112 percent of the current 1670 MWt licensed power level (1870 MWt). The licensee reported that no new vulnerabilities to severe accidents were found and that the change in CDF due to a bounding reactor thermal power increase of 12 percent is small.

The staff reviewed Section 10.5 "Individual Plant Evaluation" of the licensee's submittal. The staff's evaluation consists of reviewing the licensee's discussion on risk pertaining to internal events (level 1), containment analysis (level 2), internal fire events, and seismic and other external events.

5.1 Internal Events

The licensee identified four probabilistic risk assessment (PRA) attributes that could potentially be affected by the power increase of 12 percent. These included (1) initiating event frequencies, (2) success criteria, (3) component failure rates, and (4) time available to operator for mitigating an accident. Each of these attributes is discussed below.

The internal initiating events reviewed for the bounding 12-percent power increase in reactor thermal power included (1) LOCA, small, medium, large, and LOCAs outside the containment; (2) anticipated transients, turbine trip, loss of feedwater, loss of condenser vacuum, manual shutdown, MSIV closure, inadvertent open relief valve, loss of drywell cooling, loss of RBCCW, loss of instrument air, loss of a DC bus, loss of service water, and reference line leak; (3) loss of offsite power (including SBO), and (4) ATWS resulting from, MSIV closure, loss of condenser, loss of offsite power, loss of feedwater, turbine trip with bypass, turbine trip without bypass, and inadvertent safety valve operation. The licensee performed a qualitative review of the underlying contributors to these initiating events to determine the potential effects of the power uprate on the initiating event frequencies. The licensee reviewed system and scram trip set points that could most likely be affected by the power uprate and determined that the set points remained within their operating limits. Therefore, the licensee concluded that power uprate would not have a significant effect on their trip frequency. The staff finds it reasonable to assume that initiating event frequencies would not be changed as long as operating band/limits of equipment are not exceeded. Otherwise, any potential deviations in the initiating event frequencies in the future may be identified under the plant's Maintenance Rule program.

The systems success criteria credited in the PRA were analyzed based on a bounding 12-percent increase in power. The licensee determined that the success criteria for most systems did not change and remain adequate for the bounding 112-percent reactor power level. The only change was with the uprate analysis requiring two SRVs to open to avoid reactor overpressure whereas only one SRV opening was adequate for the 100-percent power level case. The licensee determined that the likelihood of all eight SRVs failing is not significantly different from the likelihood of at least seven out of eight SRVs failing, since the failure rate would be dominated by essentially the same common cause failure mechanism. Thus, the PRA model was not modified to reflect this change in the SRV overpressure protection success criteria. The licensee reported that all planned equipment changes to systems (as summarized in Exhibit D of the licensee submittal) maintain system functional capabilities within design criteria and that all planned changes to the emergency operating procedures will maintain the operator's ability to successfully prevent and mitigate accidents in accordance with applicable regulatory criteria.

To determine if power uprate would change the equipment/component failure rates, the licensee identified the most significant component failure event contributors to CDF (based on their risk achievement worth measures). The licensee determined that the operating band or limit for each of these components would not be exceeded by the effect of power uprate and thus concluded that there would be no long-term effect on component reliability as a result of power uprate. Similar to the above discussion on the effect of power uprate on initiating event frequency, the staff finds it reasonable to assume that component failure rate would not be changed as long as operating band/limits of component are not exceeded. Otherwise, any potential deviations in component failure rates in the future may be identified under the plant's Maintenance Rule program.

The licensee reviewed all accident mitigating operator actions that were credited in the PRA to determine if potential change in time available for operator action would result in significant change in risk. The determination of the time required to complete an action was determined using the staff guidance in NUREG/CR-4772 ("Accident Sequence Evaluation Program Human Reliability Analysis Procedure"). The licensee noted that the time to physically complete the actions does not change for a 12-percent power increase. However, the licensee determined that the maximum time available to an operator to complete a required action was shorter for the increased power condition. The licensee determined that when only a short time is available for an operator to diagnose the need for an action, a higher human error probability would be estimated for that action. The operator actions that were shown to be significant are as follows: manually depressurize the reactor, control feedwater after scram, initiate containment venting, initiate SBLC during ATWS scenarios, initiate reactor level control following shutdown with SBLC, initiate recovery of offsite power, initiate recovery of emergency diesel generators, and repair plant equipment within 48 hours (long-term heat removal failure scenarios).

The licensee reported that the power uprate significantly affected the time available to the operator for mitigating an accident. This, in turn, had an effect on the human reliability analysis performed to calculate the human error probabilities for the PRA. The licensee estimated that about two-thirds of the increase in CDF was attributed to high pressure core damage sequences that are characterized by the failure of high pressure injection system after a successful reactor scram and subsequent failure to depressurize the vessel to allow low pressure makeup. With no makeup, the licensee estimated that the time for the operator to initiate vessel blowdown is reduced from approximately 26 minutes to 23 minutes because of higher decay heat when the scram occurs at a bounding initial power of 112 percent compared to 100 percent. It was reported that a large portion of the CDF due to high pressure core damage sequences results from internal flood initiator events.

The remaining third of the overall increase in CDF was attributed to ATWS sequences. A major portion of the ATWS contribution is characterized by a turbine trip with turbine bypass to the main condenser. For most ATWS scenarios, feedwater would continue to operate and energy is released to the containment due to the relatively limited turbine bypass capacity at Monticello (about 15 percent bypass at 1670 MWt) without SBLC injection. For the turbine trip with turbine bypass ATWS scenario, the licensee reported that the time for the operator to initiate SBLC is reduced from approximately 21 minutes to 13 minutes. In spite of the reduction in time to perform this action, the likelihood of the operator correctly performing this action was estimated to be still high. As part of the emergency operating procedure training that is discussed above, the licensee noted that the operators are trained on this particular sequence in the classroom and at the Monticello plant simulator.

The licensee estimated that the CDF due to internally initiated events was about 1.6E-5/Year for the bounding uprated power level of 112 percent. This represents an increase of about 2.4E-6/Year from the CDF of 1.4E-5/Year for power level of 100 percent. As indicated above, this increase in CDF is dominated by time available for the operator to respond to accident scenarios. The licensee noted that despite this decrease in time availability, it is sufficient for the operators to perform the necessary actions. The licensee maintained that the operators are rigorously trained and evaluated on the symptom-based emergency operating procedures, which includes training on the Monticello plant simulator. In addition, multiple indications and alarms in the control room were cited as providing assistance in following the procedures.

Based on the reported analysis and results, the staff agrees that the resulting change in CDF (internal events) is mainly due to increase in human error rates which reflect decreased time available for accident mitigating operator actions. The staff believes that although virtually no significant change in initiating event frequencies, success criteria and component failure rates are predicted at this time, it remains to be seen whether these attributes will indeed be unaffected by uprated power level operation in the future. However, based on the information available at present, the staff believes that the reported increase in CDF (internal events) is small. Therefore, the staff considers the change in CDF for internal events due to the requested power increase by 6.3 percent to be acceptable.

5.2 Level 2 Internal Events PRA

The licensee reported that from the baseline (100-percent power level) level 2 PRA results, the potential for a large early release is small, on the order of 3 percent of the total CDF. As with other Mark I containments, large early releases for Monticello are dominated by ATWS and interfacing LOCA sequences.

For the bounding uprated power level at 112 percent, the licensee determined that the large early release frequency (LERF) was approximately 3% of CDF, the same percentage as for the baseline. Since the CDF for the uprated power level increased slightly compared to the baseline, the uprate LERF also increased slightly. The changes in the Level 2 quantification resulted from the changes made to the Level 1 accident sequence analysis due to reduced time available for operator recovery action. The ATWS sequences dominate the increase in large early releases due to the shorter time available to the operator to initiate standby liquid control. The major contributors to large early releases remained the same as in the baseline analysis and include ATWS, hydrogen combustion, and interfacing LOCA sequences. As in the base case analysis, the majority of the Level 2 accident sequences either do not result in containment failure, are vented or released through a pool, or are estimated to occur many hours into the accident. Based on the small increase in LERF, the staff considers the change in LERF due to requested power increase by 6.3 percent to be acceptable as it meets the criteria of DG-1061.

5.3 Internal Fire, Seismic, and Other External Events PRA

The CDF contribution from internal fires increased from 8.34E-6/Year to 8.8E-6/Year. This was attributed solely to the increase in human error rates because the time available to perform various accident mitigating tasks decreases with uprate. The decrease in time available with uprate is due to higher core decay heat increasing the steaming rate and thus leading to an earlier core uncovery. A majority of the change in CDF occurs due to scenarios involving core damage occurring at high pressure. This is attributed to the decrease in the time available for the operator to blow down the vessel before the core becomes uncovered. The remaining CDF increase involves sequences related to long-term containment heat removal, and a reduction in time to repair failed decay heat removal equipment (from 27 hours to 24 hours).

The licensee reported that there were no changes to the plant's capability to cope with a seismic event due to the power uprate. In addition, the potential for and capability of the plant to withstand "other" external event initiators were found to be essentially unaffected by the power uprate. A sensitivity study performed for tornado missiles showed that the difference in available operator response times resulted in a negligible change in CDF.

Based on these reported changes in CDF due to internal fire, seismic event, and other potential external initiators, the staff considers the CDF change to be small. Therefore, the staff considers the CDF change for these events due to the 6.3 percent increase in power to be acceptable.

5.4 Quality of PRA

The licensee's original IPE was submitted to the NRC in 1992 and the staff's safety evaluation accepting the submittal was issued by the staff in 1994. As stated in the safety evaluation, the staff found that (1) the IPE was complete with respect to the information requested in Generic Letter 88-20 and associated supplement 1, (2) the analytic approach was technically sound and capable of identifying plant-specific vulnerabilities, including those associated with internal flooding, (3) the licensee employed a viable means to verify that the IPE models reflect the current plant design and operation at time of submittal to the NRC, (4) the IPE had been peer reviewed, (5) the licensee participated in the IPE process, (6) the IPE specifically evaluated the decay heat removal function for vulnerabilities, and (7) the licensee responded appropriately to the Containment Performance Improvement program recommendations. Based on these findings, the staff concluded that the licensee met the intent of Generic Letter 88-20 ("Individual Plant Examination on Severe Accident Vulnerability").

The CDF reported in the original IPE was 2.6E-5/Year. The latest updated PRA (baseline) estimated the CDF at 1.37E-5/Year. The licensee attributed this decrease in CDF to changes in the model as well as improvements that have been made since the IPE. These changes include (1) diesel generator 13 backfeed through emergency bus 15 to supply battery chargers, (2) installation of the hard pipe vent which provides an additional means for containment heat removal, (3) improvements to SRV pneumatics (including power supplies), (4) diesel fire pump as an additional source of low pressure makeup water, (5) addition of air compressor 14 which is not dependent on service water, (6) success criteria for service water changing from 2 pumps to 1 pump, and (7) updated internal floods analysis.

The licensee reported that the internal events PRA used for the power uprate evaluation is based on a more current version of the PRA than the version used for the IPE. Although the licensee did not provide a full documentation of their PRA, a review of their submittal pertaining to PRA for power uprate as well as information contained in the original IPE submittal and the safety evaluation provided sufficient indication to the staff that the licensee's PRA and their analysis for power uprate are adequate to support the power uprate request.

5.5 Conclusion

The licensee reported that no new vulnerabilities to severe accidents were found due to bounding power uprate to 112 percent. Based on the reported analysis and results, the staff agrees that the resulting change in CDF (internal events) and LERF is mainly due to increase in human error rates which reflect decreased time available for accident mitigating operator actions. The staff agrees that based on current analysis, no significant change can be predicted for initiating event frequencies, success criteria and component failure rates.

Based on the information available at present, the staff believes that the reported increase in CDF (internal events) and LERF, although not insignificant, are small and that operating the plant at the requested uprated power level does not pose undue risk at the plant. The staff recognizes that these results are based on the licensee's analysis performed for a bounding power level of 112 percent. Therefore, the staff considers the change in CDF and LERF due to the requested power increase by 6.3 percent to be acceptable.

6.0 POWER ASCENSION TEST PROGRAM FOR POWER UPRATE

As stated in its letter dated July 1, 1998 (Ref. 13), the licensee has developed the uprate power ascension test program in accordance with the guidelines provided in Reference 16. The uprate power ascension test program is designed to verify the following: (1) plant systems affected by the power uprate are within design limits; (2) nuclear fuel thermal limits are maintained within expected margins; (3) the response of the main steam pressure control system is stable; (4) the response of the reactor water level control system is stable; and (5) the reactor core flow is within design limits and stable.

The uprate power ascension will consist of three power increase increments of approximately 35 MWt or 2.1 percent. The incremental increases provide for a controlled approach to the uprate power level. The power increase increments are: (1) 1670 to 1705 MWt; (2) 1705 to 1740 MWt; and (3) 1740 to 1775 MWt. Steady-state conditions will be established after each increment of power increase, and test data will be obtained to confirm the response of plant parameters and system performance. The station management approval of the test summary will be required prior to each power increment increase.

The conduct of the uprate power ascension test program will be controlled in accordance with the licensee's procedures. Corrective actions for equipment failing to meet the established acceptance criteria will be made in accordance with the procedures or TS action statements as appropriate.

7.0 EVALUATION OF CHANGES TO FACILITY OPERATING LICENSE (FOL) AND TS

As documented in the previous sections of this safety evaluation, the staff has reviewed the following FOL and TS changes requested by the licensee to support its power uprate program:

7.1 Changes to the Operating License

Operating License DPR-22, Docket No. 50-263, page 3, paragraph C.1, Maximum Power Level, states:

> The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 1670 megawatts (thermal).

Operating License DPR-22, Docket No. 50-263, page 3, paragraph C.1, Maximum Power Level, is revised to state:

> The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 1775 megawatts (thermal).

7.1.1 License Conditions

In its July 30, 1998, letter the licensee proposed the following commitments with the understanding that these commitments will become license conditions:

a) All affected environmental qualification files, including service life and maintenance intervals if necessary, shall be revised to reflect the new environmental profile changes associated with power uprate pior to implementation of the amendment (prior to exceeding 1670 MWt).

b) All affected process computer and SPDS data points shall be changed to reflect uprate operating conditions prior to implementation of the amendment (prior to exceeding 1670 MWt).

c) Control room simulator changes shall be completed in accordance with ANSI/ANS 3.5-1985 Section 5.4.1, Simulator Performance Testing, and Monticello simulator configuration control procedures prior to implementation of the amendment (prior to exceeding 1670 MWt).

d) Classroom and simulator training on new knowledge and abilities associated with the power uprate shall be provided in accordance with Monticello Training Center procedures prior to implementation of the amendment (prior to exceeding 1670 MWt).

e) NSP shall monitor plant operational parameters for uprate impacts on the PRA models during and after the power uprate ascension test program.

f) Control room simulator changes shall be verified against actual plant startup data within 3 months of completion of the power uprate ascension test program.

g) The applicable training programs and the simulator shall be modified, or appropriate compensatory actions shall be taken, in accordance with the Monticello Training Center procedures to reflect issues and discrepancies identified during startup testing within 6 months of completion of the power uprate ascension test program.

h) The MNGP USAR shall be updated within 9 months of completion of the power uprate ascension test program to reflect the changes associated with power uprate operation. This update shall not include credit for suppression pool scrubbing in the MSIV leakage pathway in the revised LOCA analysis.

i) NSP shall evaluate whether MO-2034 and MO-4229 are capable of allowing a subsequent operation after the required isolation safety functions are completed. This evaluation may include an examination of assumptions and methodologies, additional administrative controls, and modifications. The evaluation shall be completed in order to institute the corrective actions, if any, by the end of the next scheduled refueling outage from the date of this amendment.

j) NSP shall evaluate the capacity margins of MO-2398 and MO-2034. This evaluation may include an examination of assumptions and methodologies, additional administrative controls, and modifications. The evaluation shall be completed in order to institute the corrective actions by the end of the next scheduled refueling outage from the date of this amendment.

7.2 Changes to the TS and Associated TS Bases

7.2.1 TS Section 1.0, DEFINITIONS

a) The definition for Rated Neutron Flux provided in Section 1.0, page 4 of the TS states:

R. Rated flux is the neutron flux that corresponds to a steady-state power level of 1670 thermal megawatts..

The definition is revised to state:

R. Rated flux is the neutron flux that corresponds to a steady-state power level of 1775 thermal megawatts.

b) The definition for Rated Thermal Power provided in Section 1.0, page 4 of the TS states:

S. Rated thermal power means a steady-state power level of 1670 thermal megawatts.

The definition is revised to state:

S. Rated thermal power means a steady-state power level of 1775 thermal megawatts.

7.2.2 Bases for TS Section 2.3, Pages 14 and 15.

a) Revise the Bases discussion to reflect that the abnormal operational transients have been analyzed to thermal powers of 1775 MWt and to reflect the proposed change to the licensed power level of 1775 MWt on page 14 of the Bases for TS Section 2.3.

b) Revise paragraph 'A' on page 15 of the Bases for TS Section 2.3 to reflect the proposed change to the licensed power level of 1775 MWt.

7.2.3 TS Sections 2.3.A.1.a, 2.3.A.1.b, Table 3.2.3, and Associated Bases

a) TS 2.3.A.1.a and 2.3.A.1.b state:

For two recirculation loop operation (TLO):

$S \leq 0.66W + 70\%$ where

S = Setting in percent of rated thermal power, rated power being 1670 MWt.

W = Percent of the drive flow required to produce a rated core flow of 57.6×10^6 lb/hr.

For single recirculation loop operation (SLO):

$S \leq 0.58(W - 5.4) + 62\%$

The specifications are revised to state:

For two recirculation loop operation (TLO):

$S \leq 0.66W + 65.6\%$ where

S = Setting in percent of rated thermal power, rated power being 1775 MWt.

W = Percent of recirculation drive flow required to produce a core flow of 57.6×10^6 lb/hr.

For single recirculation loop operation (SLO):

$S \leq 0.66(W - 5.4) + 65.6\%$

b) TS Table 3.2.3, Function item 3, APRM, specifies the following APRM upscale trip settings:

(1) TLO, Flow Biased $\leq 0.66W + 58\%$

(2) SLO, Flow Biased $\leq 0.58(W - 5.4) + 50\%$

The specification is revised to state:

(1) TLO, Flow Biased $\leq 0.66W + 53.6\%$

(2) SLO, Flow Biased $\leq 0.66(W - 5.4) + 53.6\%$

c) Bases for TS Section 2.3 on page 16 and TS Section 3.5/4.5 on page 114 are revised to provide reference to the evaluation performed for MNGP power uprate.

7.2.4 Bases for TS Sections 2.2 (Page 23), 2.4 (Page 24), and 3.6/4.6 (Page 150)

Bases for TS Sections 2.2, 2.4, and 3.6/4.6 are revised to clarify the design basis pressurization event as it applies to SRV capacity, to clarify the reactor pressure safety limit, and to reflect the proposed change in the licensed thermal power limit to 1775 MWt.

7.2.5 TS Section 3.1 and Associated Bases

a) TS Section 3.1, TABLE 3.1.1, Trip Function item 9 on page 28 states that the Turbine Condenser Low Vacuum Limiting Trip Settings is ≥ 23 inches of mercury (23 in. Hg).

The specification is revised to state that the Turbine Condenser Low Vacuum Limiting Trip Setting is ≥ 22 inches of mercury (22 in. Hg).

b) Bases for TS Section 3.1 on page 37 is revised to reflect that scram function occurs at greater than or equal to 22 inches of mercury (22" Hg) vacuum based on MNGP power uprate evaluations.

c) TS Section 3.1, TABLE 3.1.1, item D on page 30 states:

D. Reactor power less than 45% (751.5 MWt).

The specification is revised to state:

D. Reactor power less than 45% (798.75 MWt).

d) TS Section 3.1, TABLE 3.1.1, item d on page 30 states:

d. The turbine stop valve closure and fast control valve closure scram functions when the reactor thermal power is ≤ 45% (751. 5 MWt).

The specification is revised to state:

d. The turbine stop valve closure and fast control valve closure scram functions when the reactor thermal power is ≤ 45% (798.75 MWt).

e) Bases for TS Section 2.3 on page 19 and TS Section 3.1 on page 38, concerning the discussion of plant conditions for which the turbine control valve fast closure scram and turbine stop valve scram may be bypassed, are revised to reflect the plant conditions for 1775 MWt.

f) TS Section 3.1, TABLE 3.1.1, Function 7 on page 28 states:

Reactor Low Water Level ≥ 7 in. (6)

where note 6 on page 30 states that 7" water level instrumentation is 10'6" above the top of the active fuel at rated power.

This specification is revised to state:

Reactor Low Water Level ≥ 7 in. (annulus)

and note 6 on page 30 is deleted.

7.2.6 TS Section 3.2 and Associated Bases

a) TS Section 3.2, TABLE 3.2.1, Function 3.a on page 50 states:

Low Reactor Water Level ≥ 10'6" above the top of the active fuel

This specification is revised to state:

Reactor Low Water Level ≥ 7" (annulus)

b) Bases for TS Section 3.2 on page 64 are revised to include discussion of reactor water level changes due to higher pressure drop across the dryer/separator at the uprated operating conditions.

c) TS Section 3.2, TABLE 3.2.1, Function 6.a on page 50 states:

Reactor Pressure Interlock ≤ 75 psig at pump suction

This specification is revised to state:

Reactor Pressure Interlock ≤ 75 psig at the reactor steam dome

d) TS Section 3.2, Table 3.2.2, Function C.3 on page 53 states:

Low Pressure Core Cooling Pumps Discharge Pressure Interlock ≤ 100 psig

This specification is revised to state:

Low Pressure Core Cooling Pumps Discharge Pressure Interlock ≥ 60 psig ≤ 100 psig

e) Bases for TS Section 3.2 on page 65 are revised to reflect the above changes to TS Section 3.2.

7.2.6 TS Section 3.5.C and Associated Bases

TS Section 3.5.C, Containment Spray/Cooling System, on page 104 states:

C. Containment Spray/Cooling System

1. Except as specified in 3.5.C.2, 3 and 4 below, both Containment Spray/Cooling Subsystems shall be operable whenever irradiated fuel is in the reactor vessel and reactor water temperature is greater than 212 °F. A containment/spray cooling subsystem consists of the following equipment powered from one division:

   2 RHR Service Water Pumps
   1 Heat Exchanger
   2 RHR Pumps*
   Valves and piping necessary for:
      Torus Cooling
      Drywell Spray

2. One RHR Service Water Pump may be inoperable for 30 days.

3. One RHR Service Water Pump in each subsystem may be inoperable for 7 days.

4. One Containment Spray/Cooling Subsystem may be inoperable for 7 days.

5. If the requirements of 3.5.C. 1, 2, 3 and 4 cannot be met, an orderly shutdown of the reactor will be initiated and the reactor water temperature shall be reduced to less then 212 °F within 24 hours.

* For allowed out of service times for the RHR pumps see Section 3.5.A.

The specification is revised to state:

C. Containment Spray/Cooling System

1. Except as specified in 3.5.C.2 below, both Containment Spray/Cooling Subsystems shall be operable whenever irradiated fuel is in the reactor vessel and reactor water temperature is greater than 212 °F. A containment/spray cooling subsystem consists of the following equipment powered from one division:

   1 RHR Service Water Pump
   1 RHR Heat Exchanger
   1 RHR Pump*
   Valves and piping necessary for:
      Torus Cooling
      Drywell Spray

2. One Containment Spray/Cooling Subsystem may be inoperable for 7 days.

3. If the requirements of 3.5.C.1 or 2 cannot be met, an orderly shutdown of the reactor will be initiated and the reactor water temperature shall be reduced to less than 212 °F within 24 hours.

* For allowed out of service times for the RHR pumps see Section 3.5.A.

The above change is based on a previously issued safety evaluation (Ref. 23), and therefore the staff finds it acceptable. The licensee should revise Bases for TS Section 3.5.C to reflect the above changes to the TS.

7.2.7 Administrative/Editorial Changes

The licensee has proposed the following editorial changes:

a) TS, Table of Contents, Page ii - The subject headings for subsections A, B, and C of Section 3.4 and 4.4, Standby Liquid Control System, are revised to reflect the headings in the body of the specifications. Heading A is changed from "Normal Operation" to "System Operation." Heading B is changed from "Operation with Inoperable Components" to "Boron Solution Requirements." Heading C, which states "Volume Concentration Requirements" is changed such that the phrase "Volume Concentration Requirements" is deleted as subsection C has no heading in the body of the specifications.

b) TS, Table of Contents, Page ii - The subject heading for subsection A of Section 3.5 and 4.5, Core and Containment/Spray Cooling Systems is revised to reflect the heading in the body of the specifications. Heading A is changed from "ECCS" to "ECCS Systems."

c) TS, Table of Contents, Page ii - Page listings for various subsections are revised to reflect actual page numbering in the body of the specifications. The page numbers listed for 3.5/4.5 Bases is changed from 109 to 110; 3.6 and 4.6 Bases is changed from 144 to 145; subsection F, Deleted, of section 3.6 and 4.6, Primary System Boundary, is added as page 128; and subsection E, Combustible Gas Control System, of section 3.7 and 4.7, Containment Systems, is changed from 171a to 172

d) TS, Table of Contents, Page iii - Page listings for various subsections are revised to reflect actual page numbering in the body of the specifications. The page numbers listed for subsection 4, Station Battery System, of Section 3.9 and 4.9, Auxiliary Electrical System, is changed from 202 to 203, and 4.11, Bases, is changed from 217 to 218.

e) TS, Table of Contents, Page iii - Subsection 5, 24V Battery Systems, is added under Section 3.9 and 4.9, Auxiliary Electrical Systems, to reflect the body of the specifications.

f) Bases for TS Section 3.5/4.5 - Revise the last sentence of the Section A, ECCS Systems, on page 112 concerning the allowed out-of-service time for the selected SRVs which form part of the ADS. The Bases currently states an incorrect allowed out-of-service time of seven (7) days. The bases is revised to be consistent with TS 3.5.A.3.h which establishes an allowed out-of-service time of fourteen (14) days.

g) Bases for TS Section 3.5/4.5 - The Bases discussion for the RHR intertie line is revised to clarify the use and purpose of the intertie line.

h) Bases for TS Section 3.1 - The Bases were revised to include a statement on the application of GE setpoint methodology.

i) Bases for TS Section 3.2 - Delete first two sentences in the last paragraph.

The staff has reviewed the above proposed changes to FOL and TS to support operation at the increased power level and finds these changes acceptable since these changes will not adversely impact the safe operation of the plant at the increased power level and that there is a reasonable assurance that the radiological consequences of normal operations and anticipated accidents at the increased power level will continue to meet the applicable regulatory requirements.

8.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Minnesota State official was notified of the proposed issuance of the amendment. The State official had no comments.

9.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, 51.33, and 51.35, and the staff's February 8, 1996, position paper, a draft environmental assessment and finding of no significant impact on the proposed amendment was published in the *Federal Register* for a 30-day comment period (January 27, 1998, 63 *FR* 3929). There were no comments on the proposed action. Accordingly, based upon the environmental assessment and final finding of no significant impact, the Commission has determined that issuance of this amendment will not have a significant effect on the quality of the human environment (63 *FR* 46489).

10.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors:

Principal Contributors:	R. Frahm, Sr.	J. Lee
	A. Ulses	L. Brown
	S. Lee	M. Mitchell
	R. Goel	H. Garg
	C. Wu	V. Ordaz
	A. Lee	J. Ma
	R. Rothman	C. Craig
	J. Knox	G. West
	T. Kim	T. Scarbrough

Attachments: 1. References
2. List of Abbreviations

Date: September 16, 1998

REFERENCES

1. Letter from W. J. Hill (Northern States Power Company (NSP)) to NRC, "License Amendment Request Dated July 26, 1996 Supporting the Monticello Nuclear Generating Plant, Power Uprate Operation," dated July 26, 1996. (With attachments. Exhibits E and G proprietary. Not publicly available.)

2. Letter from W. J. Hill, (NSP) to NRC, "Response to Request for Additional Information (RAI) on Monticello Nuclear Generating Plant Power Rerate Program (TAC No. M96238)," dated September 5, 1997. (With attachments.)

3. Letter from M. F. Hammer (NSP) to NRC, "Revision 1 to License Amendment Request Dated July 26, 1996 Supporting Monticello Nuclear Generating Plant Power Rerate Program," dated December 4, 1997. (With attachments. Proprietary and nonproprietary documents available.)

4. Letter from M. F. Hammer (NSP) to NRC, "Response to February 11, 1998 Request for Additional Information on Monticello Power Rerate License Amendment (TAC No. M96238)," dated March 6, 1998. (With attachments.)

5. Letter from M. F. Hammer (NSP) to NRC, " NSP Response to Supplemental Request for Additional Information Concerning the Monticello Nuclear Generating Plant Power Rerate Program (TAC No. M96238)," dated March 26, 1998. (With attachments. (Portion of Att. 4 proprietary. Not publicly available.)

6. Letter from M. F. Hammer (NSP) to NRC, "Submittal of Supplemental Information and Identification of Commitments for Monticello Power Rerate Program," dated April 8, 1998. (With attachments.)

7. Letter from M. F. Hammer (NSP) to NRC, "Submittal of Information Regarding the Seismic Verification of the MSIV Leakage Path at Monticello (TAC No. M96238)," dated April 17, 1998.

8. Letter from M. F. Hammer (NSP) to NRC, "Submittal of Additional Information Regarding Monticello Power Rerate Program (TAC No. M96238)," dated April 22, 1998.

9. Letter from M. F. Hammer (NSP) to NRC, "Supplemental Information in Regard to Monticellot Power Rerate License Amendment (TAC No. M96238)," dated May 5, 1998. (Att. 2 proprietary. Not publicly available.)

10. Letter from M. F. Hammer (NSP) to NRC, "Response to April 24, 1998 Request for Additional Information (RAI) on Monticello Power Rerate License Amendment (TAC No. M96238)," dated May 12, 1998.

11. Letter from M. F. Hammer (NSP) to NRC, "Demonstration of the Seismic Qualification of the MSIV Leakage Path at Monticello (TAC No. M96238)," dated May 29, 1998.

12. Letter from M. F. Hammer (NSP) to NRC, "Seismic Verification of the MSIV Leakage Path at Monticello (TAC No. M96238)," dated June 15, 1998.

13. Letter from M. F. Hammer (NSP) to NRC, "Supplemental Information Regarding the Monticello Power Rerate (TAC No. M96238)," dated July 1, 1998.

14. Letter from M. F. Hammer to NRC, "Supplemental Information Regarding the Seismic Verification of the MSIV Leakage Path at Monticello (TAC No. M96238)," dated July 20, 1998.

15. Letter from M. F. Hammer (NSP) to NRC, "Supplementary Information Regarding the Monticello Power Rerate (TAC No. M96238)," dated July 30, 1998.

16. General Electric Company (GE), Licensing Topical Report NEDC-32424P, "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate" (ELTR1), dated February 1995. (Transmitted to NRC by letter from K.K. Berry (GE) dated February 24, 1995.) (Proprietary and nonproprietary reports available.)

17. Letter from D. M. Crutchfield (NRC) to G. L. Sozzi (GE), "Staff Position Concerning General Electric Boiling Water Reactor Extended Power Uprate Program, TAC No. M91680," dated February 8, 1996.

18. GE, Licensing Topical Report No. NEDC-32523P, "Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate" (ELTR2), dated March 1996 (transmitted by letter from W. Marquino (GE) to NRC dated March 22, 1996) (proprietary and nonproprietary reports available); and Supplement 1, Volumes 1 and 2, dated June 1996 (transmitted by letter from W. Marquino (GE) to NRC dated July 8, 1996) (proprietary information; not publicly available).

19. Letter from T. H. Essig (NRC) to J. Quirk (GE), "Staff Safety Evaluation of General Electric Boiling Water Reactor (BWR) Extended Power Uprate (TAC M95087)," dated September 14, 1998.

20. USNRC, SECY-97-042, "Response to OIG Event Inquiry 96-04S Regarding Maine Yankee," dated February 18, 1997.

21. GE, Licensing Topical Report NEDC-32546P, "Power Rerate Safety Analysis Report for Monticello Nuclear Generating Plant," Revision 0 dated July 1996 (proprietary information; not publicly available), and Revision 1, dated December 1997 (proprietary and nonproprietary reports available). (Transmitted by References 1 and 3, respectively.)

22. Letter from W. J. Hill (NSP) to NRC, "Update of Design Basis Accident Containment Temperature and Pressure Response (TAC No. M97781)," dated June 19, 1997. (With attachments.)

23. USNRC, "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 98, Facility Operating License No. DPR-22, Northern States Power

Company, Monticello Nuclear Generating Plant, Docket No. 50-263," dated July 25, 1997. (Transmitted by letter from T. Kim (NRC) to R.O. Anderston (NSP), "Issuance of Amendment Re: Updated Analysis of DBA Containment Temperature and Pressure Response and Reliance on Containment Pressure to Compensate for Potential Deficiency in NPSH for ECCS Pumps During DBA.)

24. GE, Licensing Topical Report NEDE-23785-PA, "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident,"June 1984. (Proprietary information. Not publicly available.)

25. GE, Licensing Topical Report, NEDC-32514P, "Monticello SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," dated July 1996. (Proprietary information. Not publicly available.) (Transmitted by Reference 1.)

26. GE, Licensing Topical Report, NEDC-32514P, Rev. 1, "Monticello SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," dated December 1997. (Proprietary and nonproprietary documents available.) (Transmitted by Reference 3.)

27. Letter from W. J. Hill (NSP) to NRC, "Reactor Coolant Equivalent Radioiodine Concentration and Control Room Habitability (TAC No. M962556)," dated July 26, 1996, and Revision 1 dated April 11, 1997.

28. USNRC, NUREG/CR-5055, "Atmospheric Diffusion for Control Room Habitability Assessments," (prepared by J.V. Ramsdell, Battelle Memorial Institute, Pacific Northwest Laboratory), May 1988.

29. USNRC, NUREG/CR-6331, Rev. 1, "Atmospheric Relative Concentrations in Building Wakes," (prepared by J. V. Ramsdell, et al., Battelle Memorial Institute, Pacific Northwest National Laboratory, May 1997.

30. Letter from M. F. Hammer (NSP) to the NRC, "Supplemental Information in Regard to Monticello Power Rerate License Amendment (TAC No. M96238)," dated May 5, 1998. (Attachment 1 to Attachment 2 proprietary. Not publicly available.)

31. GE Nuclear Energy,"BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems," NEDC-31858P, Rev. 2, dated September 1993. (Proprietary information. Not publicly available.)

32. "Supplemental Safety Evaluation Report No. 2 on Seismic Qualification Utility Group's Generic Implementation Procedure, Revision 2, Corrected February 14, 1992."

33. Memorandum from Yi-Ben Tsai (PG&E Geosciences) to Nicholas J. Markevich (PG&E Civil Engineering), dated August 12, 1992.

34. "Monticello Nuclear Generating Plant Verification of Seismic Adequacy of Mechanical and Electrical Equipment, Unresolved Safety Issue A-46 (SQUG)," Northern States Power Company, November 1995.

35. USNRC, "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 101, Facility Operating License No. DPR-22, Northern States Power Company, Monticello Nuclear Generating Plant, Docket No. 50-263," dated August 28, 1998. (Transmitted by letter from T. Kim (NRC) to R.O. Anderson (NSP), "Issuance of Amendment Re: Reactor Coolant Equivalent Radioiodine Concentration and Control Room Habitability (TAC NO. M96256).

LIST OF ABBREVIATIONS

ADS	automatic depressurization system
AISC	Association of Iron and Steel Engineers
AOV	air-operated valve
APRM	average power range monitor
ART	adjusted reference temperature
ASME	American Society of Mechanical Engineers
ATWS	anticipated transients without scram
BOP	balance of plant
BWR	boiling-water reactor
BWROG	BWR Owners' Group
BWRVIP	Boiling Water Reactor Vessel and Internals Project
CDF	core damage frequency
CGCS	combustible gas control system
CPR	critical power ratio
CRD	control rod drive
CRDM	control rod drive mechanium
CS	core spray
CUF	cumulative fatigue usage factor
DBA	design-basis accident
E/C	erosion/corrosion
EAB	exclusion area boundary
ECCS	emergency core cooling system
EDG-ESW	emergency diesel generator-emergency service water
EOL	end of license
EPRI	Electric Power Research Institute
EQ	environmental qualification
EQE	Earthquake Engineering, Inc.
ESW	emergency service water
FOL	facility operating license
FRS	floor response spectrum
FSAR	final safety analysis report
GE	General Electric Company
GIP	generic implementation procedure
HELB	high-energy line break
HP	high pressure
HPCI	high-pressure coolant injection
HVAC	heating, ventilation, and air conditioning
ICF	increased core flow
IPE	individual plant examination
LBPCT	licensing basis peak cladding temperature
LERF	large early release frequency
LOCA	loss-of-coolant accident
LOFW	loss of feedwater
LP	low pressure
LPCI	low-pressure coolant injection
LPZ	low-population zone
LTR	licensing topical report
LVSSR	lateral-to-vertical support span ratio
MAPLHGR	maximum average planar linear heat-generation rate

MELLL	maximum extended load line limit
MELLLA	maximum extended load line limit analysis
MEOD	maximum extended operating domain
MNGP	Monticello Nuclear Generating Plant
NOMPCT	nominal peak cladding temperature
MOV	motor-operated valve
MSIV	main steam isolation valve
MSLBA	main steam line break accident
NPDES	National Pollutant Discharge Elimination System
NPSH	net positive suction head
NSP	Northern States Power Company
NSSS	nuclear steam supply system
OLMCPR	operating limit minimum critical power ratio
P-T	pressure-temerature
PCT	peak cladding temperature
PGA	peak ground acceleration
PG&E	Pacific Gas and Electric Company
PLHGR	peak linear heat-generation rate
PRA	probabilistic risk assessment
RBCCW	reactor building closed-cooling water
RBM	rod-block monitor
RCIC	reactor core isolation cooling
RCPB	reactor coolant pressure boundary
RHR	residual heat removal
RHRSW	residual heat removal service water
RIPD	reactor internal pressure difference
RPV	reactor pressure vessel
RWCU	reactor water clean-up
SAM	seismic anchor motion
SBA	small-break accident
SBLC	standby liquid control
SBO	station blackout
SFPCS	spent fuel pool cooling system
SGTS	standby gas treatment system
SLCS	standby liquid control system
SLO	single-loop operation
SPDS	safety parameter display system
SQUG	Seismic Qualification Utility Group
SRP	Standard Review Plan
SRV	safety/relief valve
SSC	structure, system, and component
SSE	safe-shutdown earthquake
TLO	two-loop operation
TS	technical specification
UBPCT	upper bound peak cladding temperature
UHS	ultimate Heat sink
USAR	updated safety analysis report
USE	upper-shelf energy
USGS	United States Geological Survey
USI	unresolved safety issue
VSR	vertical support span ratio